Generation IV Roadmap

Crosscutting Fuel Cycle R&D Scope Report

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Background

The Charter of the Generation IV Roadmap Fuel Cycle Crosscut Group (FCCG) is to (1) examine the fuel cycle implications for alternative nuclear power scenarios in terms of Generation IV goals and (2) identify key fuel cycle issues associated with Generation IV goals. This included examination of "fuel resource inputs and waste outputs for the range of potential Generation IV fuel cycles, consistent with projected energy demand scenarios." This report summarizes the results of the studies.

The membership of the FCCG comprised eight U.S. members and seven members from Generation IV International Forum (GIF) countries including members from the Organization for Economic Cooperation and Development-Nuclear Energy Agency (OECD-NEA), the International Atomic Energy Agency (IAEA), and the European Commission observer organizations to the GIF. Members of the FCCG were, in general, drawn from the Technical Working Groups (TWGs) and the Evaluation Methodology Group (EMG) of the Generation IV Roadmap organization. Five one-day working meetings were held between February 2001 and August 2001—three of them in conjunction with Generation IV TWG quarterly meetings.

The FCCG reviewed energy projections and selected the authoritative International Institute for Applied System Analysis/World Energy Council, LLC (IIASA/WEC) projections of 1998 as the basis for performing a selected set of 100-year nuclear energy futures scenarios. We reviewed the uranium ore resource projections of the OECD-NEA, IAEA, and Uranium Institute, the thorium ore resource projections from multiple sources, and investigated independent models for prediction of new ore discoveries vs. cost of supply. A survey of Generation IV concept submittals was made to define the scope of proposed fuel cycles and fuel compositions. Fuel cycle infrastructure status was reviewed, and an extensive review was undertaken of fuel cycle R&D programs underway worldwide. Documentation of the status of institutional aspects providing the enabling legal basis and boundary conditions for worldwide fuel cycle deployment was made.

On the basis of these reviews, studies and evaluations, the FCCG has produced a set of principal findings and has generated a set of top-level recommendations for Generation IV fuel cycle crosscutting R&D.

The FCCG's principal findings are based on two primary sources. First are the results from dynamic scenario simulations of various potential nuclear futures—driven by the 100-year world energy demand projections (and nuclear's share) provided by the 1998 IIASA/WEC. These nuclear futures scenarios were organized by generic fuel cycle type (once-through, partial recycle, full fissile recycle, and full transuranic recycle) and were constrained only by physically achievable mass flows and lag times of potential Generation IV power plant and fuel cycle concepts. They modeled idealized transitions from current and near-term deployments to Generation IV fuel cycles and power plants and potential symbiosis of mass flow exchanges among Generation IV power plant concept types. These scenarios provide cornerstone indicators for the Roadmap of physically achievable performance against Generation IV goals.

The second principal input to the FCCG's findings is derived from an extensive and deep review of the technical status of fuel cycle technologies deployed and under development worldwide, and an evaluation of the underlying rationale for the choices of research focus that drive these development programs. While the technical approaches vary, it was found that the worldwide underlying motivations are closely aligned to the goals articulated for Generation IV in the areas of Sustainability, Safety and

Reliability, and Economics. The fuel cycle plays a primary role in meeting the three elements of the Generation IV sustainability goals.

The principles of sustainability include meeting society's needs for energy services while using the earth's resources in an efficient and environmentally friendly way. Nuclear fission converts uranium and thorium resources to energy with fission products as the essential waste. The net production of long-lived transuranium isotopes is a characteristic of the specific reactor types and fuel recycling steps used. The goals of Generation IV include reduced waste generation and more efficient use of ore resources along with making the nuclear fuel cycle the least attractive route to proliferation of nuclear armaments.

Today, the cost of uranium and thorium is not a major contributor to the cost of nuclear energy, and resources do not constrain the expansion of nuclear power. Within several decades the costs of fuel materials may become more significant as lower-grade resources are used. However, repository capacity is an increasingly expensive and politically divisive constraint on growth of nuclear power. The use of fuel cycles and reactors that minimize repository requirements is essential to increased use of nuclear energy.

Principal Findings

Reversing Waste Buildup in a Growing Nuclear Economy

Closed fuel cycles have already demonstrated a significant reduction of the volume and long-term radiotoxicity of nuclear high-level waste through the reprocessing of 20,000 tonnes of spent light water reactor (LWR) fuel to recycle the plutonium and uranium. Closed fuel cycles provide the opportunity to partition classes of nuclear waste and to manage each class in a separate waste form according to its individual characteristics. Advanced waste management strategies include transmutation of selected nuclides, cost-effective decay heat management, flexible interim storage, and customized waste forms for specific geologic repository environments. These strategies hold the promise to significantly reduce the long-lived radiotoxicity of the waste destined for geological repositories by at least an order of magnitude via recovery of virtually all the heavy long-lived radioactive elements. Such reductions, and the ability to optimally condition the residual wastes and manage heat loads, will permit far more efficient use of limited repository capacity and further enhance overall safety of the final disposal of radioactive wastes. An equivalent reduction in secondary waste arisings is also possible.

Advanced once-through cycles also have a potential to provide useful improvements in repository performance, although smaller than closed cycles. These improvements may be achieved primarily through the increase of thermal efficiency and high levels of fuel burn up.

Sustainable Use of Resources

Virgin uranium ore supplies are ensured for several decades, but will fall short of demand for low-cost uranium by the middle of the 21st century. Timely renewal of exploration campaigns may further extend this virgin uranium supply but probably at a higher cost. However, more efficient use of raw materials is sustainable while reducing the environmental burdens and worker radiological exposures from mining and milling activities. More efficient use of ore can be achieved in three ways: by further extraction of fuel from existing stocks of depleted uranium, by more fuel-efficient reactors, or by recycle of existing stocks of discharged fuel. (Additionally, down blending of stockpiles of highly enriched uranium declared to be in excess of security needs and the use of stored uranium inventories will be temporary sources of fuel for the near-term.)

Nuclear energy has the important market advantage that its fuel cycle contributes only about 20% to the overall cost of energy. This advantage provides remarkable flexibility for decoupling the strategies for meeting Generation IV economics and safety and reliability goals from the strategies for meeting sustainability goals. Wide ranges of fuel cycle approaches can be undertaken with only weak influence on economics. Power plant design strategies to meet economic and safety and reliability goals, while intimately tied to fuel design, are only loosely influenced by fuel cycle choices. A symbiotic mix of reactor types further enhances this flexibility.

Transition To Sustainability

Energy use is expected to grow substantially in this century; consequently, nuclear energy deployment is projected by authoritative studies to increase by as much as a factor of five or six by 2050. Over this period, the nuclear deployment will require an evolving mix of reactor types (thermal and fast spectrum) and fuel cycles (once-through and recycle) to serve different market sectors. Symbiotic mixes of reactor types and fuel cycles are essential for economically supplying the required energy, while minimizing waste generation and ensuring efficient use of resources, including limited repository capacity.

New reactor types with favorable neutron economy and a variety of fuel compositions will be required to create a symbiotic mix. These reactors can either produce fuel by breeding or consume nuclear waste by burning—providing the flexibility needed to accommodate economic and social changes. Such reactors—e.g., those employing some combination of fast neutron spectra, altered fertile (U-238, Th-232) feed, reduced neutron losses, and/or an external neutron source—rely on fuel recycle to fulfill their function.

Near-term evolutionary modifications of current reactors are needed to fill expanding energy needs while at the same time employing limited repository space with high efficiency. Within the symbiotic mix envisioned for the future, these reactors need not necessarily rely on recycle, but further increases in fuel burn up and sufficient flexibility to be adapted to a future closed fuel cycle are desirable.

Technologies for co-recycle of minor actinides with plutonium and/or for provisions of some residual contamination of recycle fuel from fission products are under current development. By focusing on creating clean waste streams (containing only the fission products), these technologies can significantly reduce the quantities of long-lived radionuclides consigned to waste and provide an intrinsic barrier to weapons proliferation. Recycle of these advanced fuels requires the handling of highly radioactive materials in all fabrication activities. Such highly radioactive and fission-product-contaminated fuels have traditionally been destined for use in the favorable neutron balance of fast reactors. However, in the symbiotic mix of reactor types envisioned for the future, such fuels may be destined for thermal reactors as well. Generation IV R&D must determine the optimal implementation of these options and develop such enabling technologies as remote fabrication for thermal reactor fuel as well as fast reactor fuel.

Fuel Cycle Safety

The radiological exposure of workers in the overall fuel cycle includes contributions from mining/milling and from recycling facilities. These worker exposures are limited and are comparable to those from reactor operation. Developments to further reduce radiological impacts, for example by use of already-mined resources and by cost-effective remote re-fabrication, will become increasingly important as the scale of nuclear power increases.

The glass waste form generated by today's fuel cycles has been shown experimentally to be very durable in repository conditions. The long-term behavior of waste forms and their ability to confine potentially mobile radionuclides are important and discriminating issues for comparing different fuel cycle strategies.

Thorium

Thorium-fueled thermal reactors and uranium fast spectrum reactors may become an attractive option in the longer term because of the depletion of uranium resources. In addition, it is believed that the joint use of thorium and uranium cycles could lead to significant reductions of the long-term radiotoxicity of the ultimate waste because of greatly reduced production of transuranium actinides. In addition, the joint use of both cycles might enhance proliferation resistance through increased U-232 and Pu-238 content in recycle fuel feedstock.

Fuel Cycle Crosscut Group's Recommendations for Fuel Cycle Crosscutting R&D for Generation IV Systems

Results of the scenarios studied by the FCCG indicate that the full fissile recycle and especially the full transuranic recycle generic fuel cycles may be needed to achieve Gen-IV sustainability goals, SU-1 and SU-2. Once-through power plant concepts, eventually symbiotically fueled with fissile material fed back from full recycle fuel cycles, will also be part of an overall sustainable nuclear energy park. Partial recycle fuel cycles can function as a bridge from the current situation to the long-term full recycle (sustainable) cycles. These observations, combined with the FCCG's conclusion regarding the likely technical achievability of full recycle (which is based on the current state of knowledge and ongoing R&D), have influenced the selection of Gen-IV fuel cycle crosscutting R&D recommended below.

Research and development targeted to "breakthrough" enabling technologies have been favored by the FCCG over those directed toward refined understanding and/or incremental improvements in existing technology. While this latter type of "incremental" R&D is essential and important for the continual improvement against Gen-IV economic, reliability and safety goals for the global nuclear energy park, it is the "breakthrough" fuel cycle enabling technologies that are needed to attain the Gen-IV waste management and resource utilization sustainability goals for the fuel cycle; these are the special domain of the fuel cycle crosscut R&D recommendations.

1. Improvement of fuels and development of advanced fuels is important no matter what fuel cycle is used. The fuel assemblies comprise the essential interface between the nuclear power plant and the fuel cycle. Fuel assemblies that achieve high discharge burn up, low reactivity loss with burn up, low fabrication cost, and which can operate at high coolant temperature for improved station thermal efficiency will directly impact economic and sustainability goals and indirectly impact safety goals. If the fuel is intended to be used in a recycle-based fuel cycle, it must be designed with ease of recycle in mind; if it is intended for once-through, it must be designed for very high discharge burn up and for extreme robustness over geologic time scales in a repository environment. Finally, because ultimately the source of fissile will be fed back from full plutonium, uranium-233, or full transuranic recycle sources, the fuel must be remotely fabricable.

    Development of new fuel types and fuel compositions containing increased plutonium and minor actinide fractions is recognized to be a significant challenge. Fuel-development campaigns typically require 10 to 15 years of significant development cost—requiring both in pile and out of pile testing (including safety tests) before the fuel is ready for deployment. While the cost is not to be understated, neither is the payoff; the fuel is the essential link between the fuel cycle (controlling resource and waste performance) and the reactor power plant (controlling cost,

reliability, and safety) against Gen-IV goals. Uranium oxide fuel is highly refined and dominates the current once-through commercial fuel cycle and is qualified for use in fast neutron spectrum reactors. Nonetheless, new Gen-IV reactor concepts and fuel cycles must be afforded opportunities to explore and exploit the potential of additional fuel types, and the R&D cost of doing so must be a central element of the Gen-IV research program.

2. Cost-effective advanced recycle technologies integrated with remote fuel re-fabrication technologies are the key enabling technologies for achieving the Gen-IV sustainability goals. In the full recycle fuel cycles, the waste destined for the repository arises from losses in recycle/re-fabrication; these losses must therefore be small. Recycle technologies that achieve "clean waste" (i.e., fission products only) and recycle "all actinide fuel" (i.e., all transuranics and optionally some fission products) back to the nuclear power plants are favored; they extract the maximum energy from the ore resource, and they consign only the fission products to the waste. Such recycle technologies based on both aqueous and on dry processes are under active development worldwide. They are yet to be developed to the prototype and commercial scales, and their continued development for cost effectiveness and low losses must be one of the cornerstones of the Gen-IV R&D program.

3. When "all actinide fuel" provides the feedstock to the fuel fabrication link in the fuel cycle, remote fabrication technologies are required. Simple, few-step processes are favored and robust fuel form designs that minimize reject rates and rework are needed. Collocation of recycle and re-fabrication facilities is favored to minimize shipping and handling of radioactive materials in bulk form.

    "All actinide fuel" is readily useable in fast spectrum reactors. However, the future global nuclear energy park is envisioned to rely on symbiosis of both fast and thermal reactor power plants, with recycled fuel used in both thermal spectrum reactors and fast reactors. Therefore, the recycle, the fuel design, and the reactor core design must all be coordinated such that thermal reactor neutronics performance is not excessively spoiled by contaminants in the fed back fuel feedstock. Research and development on this coordination, in concert with development of appropriate symbiotic mixes of reactor and fuel cycle types in the global energy park, should be a central theme of Gen-IV R&D.

4. While the importance of near-term siting, licensing, and operating a geologic repository designed for the once-through fuel cycle cannot be overstated, for the future, the development of recycle-based fuel cycles must go forward in close linkage with the development of repository designs that exploit new opportunities for customized waste forms and optimized decay heat management strategies to extend their capacities and reduce the level of stewardship required. Specifically, a singular importance in all fuel cycles rests on decay heat management for extending geologic repository capacity.

    Repository siting is and will always be difficult, so mechanisms for extending the capacity of given sites can significantly advance economic and social acceptance goals for nuclear energy. Capacity is controlled not by mass or volume, but by heat load, and heat load is dominated by Cs and Sr in the first hundred years and by minor actinides thereafter. Three alternative approaches are available for preparation of the actual waste: (1) interim storage of spent fuel prior to geologic emplacement; (2) partitioning/conditioning with interim storage of tailored waste forms prior to geologic emplacement; and (3) partitioning and interim storage of Cs/Sr with reactor recycle for consumption by fission of actinides. Design of repositories tailored for heat management and tightly integrated with the above approaches for waste preparation should be a priority development goal for Gen-IV R&D.

5.   Recycle and return to the reactor power plants of commixed plutonium and minor actinide-bearing (i.e., "all actinide") fuel creates an intrinsic radiation barrier to theft and diversion from the commercial fuel cycle; the presence of minor actinides and carryover fission products makes the recycled materials less attractive for weapons use as well as less accessible to theft or diversion. It also avoids consigning inventories of weapons-usable materials to interim storage and to geologic repositories, where over time their intrinsic protective radiation barrier decays away. On the other hand, the requirement for extrinsic measures—materials control and accounting, physical protection to detect and prevent efforts at theft, and international safeguards to detect efforts at diversion—cannot be eliminated. Research and development will be required to adapt existing regimes of extrinsic measures—which have been implemented at commercial scale for the once-through and partial recycle fuel cycles—to applicability to the full recycle fuel cycles. This R&D must be closely integrated with the development of the recycle/re-fabrication technologies and with the design of the facilities to execute these technologies.

Crosscutting Fuel Cycle R&D Scope Report

1. INTRODUCTION

The Fuel Cycle Crosscut Group (FCCG) issued its full report in draft on November 1, 2001, and in final form on March 18, 2002 (see Document 14, *Fuel Cycle Assessment Report,* on this CD). The overall fuel cycle strategy for a Generation-IV (Gen-IV) sustainable global nuclear energy infrastructure and the time phasing for its emplacement over a multiple decade transition period are discussed there. Fuel cycle crosscutting research and development (R&D) needed to achieve the Gen-IV sustainability goals are identified, justified, prioritized, and recommended in that report. The principal findings and R&D recommendations at a top level are reiterated here (in Section II) where the Executive Summary from the full report is reproduced. Because these recommendations were available (in draft form) prior to the Technical Working Groups' (TWGs) initiation of effort on the R&D scope definition, the TWGs were able to draw on those recommendations.

Therefore, concept-specific fuel cycle R&D has been identified and recommended by the TWGs. To avoid duplication, the FCCG has confined its recommendations presented here to those areas of Fuel Cycle R&D, which are complimentary to those being recommended already by the TWGs. The FCCG has concentrated here on concept-independent back end and front end links in the fuel cycle chain and on generic concept-specific fuel cycle relevant R&D, which apply across all TWGs.

As discussed in the Executive Summary from our full report, issues at the back end of the fuel cycle are more immediate in the Gen-IV multi-decade deployment sequence than those at the front end, and they are discussed first and should receive higher priority in the face of limited funding than do front end generic R&D issues. The concept-independent front end issues will become important later than do the back end issues, but will still be highly important before the mid-century time frame because forward fueling needs will cease to be ensured within currently known plus speculative reserves (as illustrated in the scenario results shown in Chapter 3 of the FCCG main report). Therefore, we enumerate the Gen-IV relevant front-end issues also—even though they are given lower priority for the viability phase of Gen-IV R&D.

As stated, the concept-specific R&D recommendations relevant to the middle links of the fuel cycle chain are being recommended primarily by the TWGs. Our R&D recommendations presented here for the middle links of the fuel cycle pertain to the generic issues of recycle/remote re-fabrication, which always arise when the all actinide recycle strategy is used.

This report stresses fuel-cycle-related R&D, which is crucial to be undertaken at an early stage in the Roadmap (i.e., during the Gen-IV Viability R&D phase). Such R&D can help inform mid-term decisions regarding which future path in overall fuel cycle approach to take at several key Roadmap branch points. Specifically, for once-through fuel cycles, it will be important to decide among several strategies on how decay heat will be managed at all times subsequent to fuel discharge from the reactor. The R&D proposed here will clarify the issues for the several choices. And separately, for recycle-based fuel cycles, it will be important to decide between a strategy of co-extraction of all transuranics versus separate extractions and potential remixing of transuranics and also between separate disposition approaches. Therefore, our R&D recommendations presented here are targeted to be performed during Viability R&D to inform strategic decisions taken at the Viability/Performance R&D transition stage during the Roadmap execution program.

These R&D recommendations are discussed in Section IV of this report. The facilities needed to carry out the research are identified in Section V.

Technology gap and recommended viability-phase R&D spreadsheets are presented in Section VI.

Although an effort has been made to complement the TWG identification of recommended R&D, the totality of fuel-cycle-related R&D will require further integration and verification of completeness during the preparation of the final Roadmap document.

2. ENUMERATION OF FUEL CYCLE R&D BEING RECOMMENDED BY THE TECHNICAL WORKING GROUPS AND ASSESSMENT OF COMPLETENESS

The FCCG R&D recommendations for the viability phase are intended to be complementary to those recommendations being made already by the TWGs. In the interest of ensuring completeness, we enumerate here what fuel-cycle-relevant R&D we understand the TWGs to be recommending:

2.1 TWG-1 Water-Cooled Reactors

The GIF-downselected water concept set is the supercritical water concept set—both thermal spectrum and (later) fast spectrum. Viability-phase R&D will be primarily for the thermal spectrum variant only. The fuel-cycle-relevant characteristics of the thermal supercritical (SC) water concept are:

Fuel:	UOX (~5% ε)
Cladding:	Stainless steel or inconel
Core Outlet Temperature:	~510ºC
Pressure:	~25–27 Mpa
Burn up:	50 MWd/kg
Cycle:	Once-through

(The longer term fast spectrum version assumes TRU-Oxide fuel in inconel cladding and multi recycle.)

No fuel-cycle-relevant R&D has been recommended by TWG-1; however, a great deal of clad/SC water compatibility R&D is stressed.

Table 1 shows a checklist of where the fuel-cycle-relevant R&D is covered for this concept set (thermal spectrum option). No holes in coverage are apparent.

2.2 TWG-2 Gas-Cooled Reactors

The GIF-downselected gas concept sets from TWG-2 are the Very High Temperature Reactor (VHTR) and the Gas-Cooled Fast Reactor.

The TWG-2 has recommended concept-specific fuel-cycle-relevant R&D for the VHTR concept operating in an open cycle. The recommendations include substantial stress on testing fuel/clad/coolant performance under all operating conditions, on fabrication technology, and on fuel performance as a geologic waste form. The R&D recommendations from the TWG leave no gap—at least for the viability phase.

The TWG-2 has recommended concept-specific fuel-cycle-relevant R&D for the Fast Gas Concept, which is intended to operate on a closed fuel cycle. The TWG's R&D recommendations are limited to a technology-based screening and selection of the basic fuel/clad concept with relevant fabrication selection and development and screening irradiation testing. As would be expected, much less focus is given to recycle/re-fabrication/waste form production R&D until the basic fuel concept options are narrowed down. During viability-phase R&D, there is not much to do beyond finding a preferred fuel concept and it is covered by the TWG recommendations.

Table 1. Supercritical water (thermal) (oxide fuel).

Fuel Cycle Link	Issues/Approach	Recommended by TWG	Recommended by TWG-3	Recommended by FCCG
1. Fuel/Clad/Coolant Compatibility Performance Normal Upset Severe Accident		Coolant/clad covered For Normal, Upset Unclear: severe	Clad/fuel covered Normal and Upset	
2. Recycle	Once-through reference			
	Options: Advanced aqueous			
	Modify front end for steel clad		Covered	
	Pyro		Covered	
	Oxide reduction			
3. Waste Form Production	Once-through reference			Covered
	Recycle options		Covered	
4. Remote Re-fabrication	N/A for reference		Covered	
	Aqueous → pellet		Covered	
	→ vibro		Covered	
	Pyro → vibro		Covered	

2.3   TWG-3 Liquid Metal-Cooled Reactors

The GIF-downselected reactor concepts from TWG-3 include Na-cooled fast reactors based on either U-TRU-Oxide fuel and advanced aqueous multi-recycle or U-TRU-Zr metal alloy fuel and pyro multi-recycle. Also included are the battery-sized concepts of Pb or Pb-Bi cooled fast reactors with pyro-based multi-recycle of the two fuel types above plus U-TRU-Nitride fuel.

All TWG-3 fuel cycle concepts presume a symbiotic tie to U oxide or MOX-fueled thermal reactors wherein the feedstock for the fast reactor cycles will be derived from discharged thermal reactor fuel.

Fuel-cycle-relevant R&D recommendations are exhaustive and complete—both for the actual fast reactor fuel cycles and for their tie to thermal reactor spent fuel as their source of feedstock. The R&D recommendations cover both advanced aqueous and pyro options (as appropriate) for oxide-based, metal-based, and nitride-based fuels. Remote re-fabrication research and development are covered for both pellet and vibrocompaction remote re-fabrication. Waste-form-development research and development are covered, as appropriate, for each recycle approach. Integrated safeguards research and development are covered for each recycle/re-fabrication approach.

2.4   TWG-4 Nonclassical Reactors

The GIF-downselected concept set from TWG-4 is the Molten Salt Reactor—with integrated fuel cycle.

For further information on the TWG-4 recommendations for fuel-cycle relevant research and development, see Document 14, *Fuel Cycle Assessment Report,* on this CD.

2.5   Assessment of R&D Recommendation Coverage

2.5.1   GIF-Downselected Concepts

In light of the TWG's coverage of the concept-specific fuel-cycle-relevant R&D for the concepts in the GIF-downselect set of concepts, the FCCG is confident that its R&D recommendations can be focused primarily on concept-independent back end and front end links of the fuel cycle chain for the GIF-downselected set of concepts.

Additionally, several truly cross-cutting R&D areas, which occur generally in the concept-specific fuel cycle R&D needs, are identified and recommended for the middle links of the fuel cycle chain in this report because the outcome of the R&D may affect fundamental branch-point decisions on the Gen-IV fuel cycle architecture:

- For once-through cycles, on how decay heat is managed—with fundamental implications for repository designs

- For recycle-based fuel cycles, on a decision between co-extraction of all transuranics versus separate extractions of transuranic elements—with fundamental implications on re-fabrication technologies.

2.5.2   Coverage of Potential Concept Additions or Evolutions

Several potential lines of development have not been captured in the GIF-downselected concepts. These include:

1.   Feedback of radioactive fuel to thermal reactors, where two issues arise—(1) adequate removal of lanthanide carryover in the recycle step and (2) remote re-fabrication of such fuel for thermal reactors

2.   Use of thorium-based fuel cycles

3.   Recycle technologies (reprocessing and remote re-fabrication) for particulate-based fuels (potentially captured in the Fast Gas Concept)

4.   Safeguards R&D for once-through fuel cycles

5.   Unanticipated innovations.

While Item 1 is strongly emphasized in the FCCG Executive Summary, and R&D recommendations and technical approaches are discussed in the main FCCG report, it has not been given priority for Viability R&D by any TWG for the GIF-downselected set of concepts because the selected concepts do not profess to recycle radioactive fuel to thermal reactors. Clearly such recycle will be essential eventually (as amply illustrated in the FCCG's scenario results presented in Chapter 3 of our main report). However, such recycle capability is not proposed for implementation right away for the GIF-selected concept set. For potential near-term implementation, these issues are already being addressed in a parallel U.S. Department of Energy (DOE) program on Partitioning and Transmutation—where the necessity for tight integration of reactor neutronics and safety design and recycle/re-fabrication optimization of lanthanide carryover is accounted for. These activities should be monitored during Gen-IV activities and should this work be terminated in the parallel program, it should most certainly become highly important to add it to the Gen-IV scope.

For Item 2, thorium-based concepts are confined to the Molten Salt Reactor (MSR) where the R&D recommendations are unique to fluid-fueled reactors.

For Item 3, the FCCG has elected to include a discussion of general aspects of reprocessing particle fuels where the challenges are severe both because of the need to "digest" the matrix to get at the fuel particles and then the need to dissolve the particles that were designed for extreme resistance to chemical attack.

Safeguards R&D must be performed integrally with recycle/re-fabrication development and such research and development are covered in the TWG-3 recommendations. Safeguards R&D must be addressed also with enrichment/direct spent fuel disposal aspects of once-through fuel cycles. The TWG-3 recommendations cover it for the recycle route, but it is covered nowhere for the once-through cycles. Research and development recommendations should perhaps await the outcome of the EMG efforts on this subject.

It is clear that revolutionary technologies, which were completely unknown or unappreciated by the Gen-IV Roadmapping activity, may come into view in the future. In the area of recycle technology, new reagents, new contacting technologies, high-temperature or plasma-related technologies, or others may emerge. Part of the role of Roadmap management must be to monitor the status of technology in related fields to identify relevant breakthroughs and bring them to bear in the Gen-IV program.

3.   FUEL CYCLE CROSSCUT GROUP RECOMMENDATIONS FOR VIABILITY-PHASE R&D COMPLEMENTARY TO TECHNICAL WORKING GROUP RECOMMENDATIONS

3.1   Concept-Independent Back End Fuel Cycle R&D

3.1.1   Systems Optimization for Integrated Fuel Cycles and Integrated Waste Management

A critical component of Gen-IV fuel cycle Viability R&D is to conduct system studies to understand and optimize waste management integrally with fuel cycle system options.

Nuclear fuel cycles have and will continue to develop incrementally as scope of deployment, societal issues, and technological knowledge change over time. It has taken four decades for the first repositories of low-level radioactive waste to become operational, and geologic repositories for high-level waste and spent nuclear fuel are expected to be operational within 1 or 2 decades more (the last step in the fuel cycle). The experience and understanding developed during a 50-year process of building the complete fuel cycle enables us now to examine and consider the entire cycle including waste management as a system rather than a set of components.

Within the Gen-IV roadmap process completed up to now, fuel cycle systems as a whole have been identified, evaluated, and analyzed by the FCCG to assist Gen-IV in the identification of preferred paths forward. However, a traditional geologic repository has always been assumed in all cases. Therefore, during Viability R&D, new and innovative waste management approaches should be identified, evaluated, and analyzed integrally with the fuel cycles to identify preferred paths forward. A set of examples can illuminate the open issues, the options, and the intent of optimization R&D.

3.1.1.1   Open Issues of Integrated Waste Management/Fuel Cycles. *Once-through fuel cycles*—The traditional once-through LWR fuel cycle is a historical accident. It was originally thought that uranium was very scarce and that all LWR spent nuclear fuel would be recycled quickly. This did not turn out to be the case for two reasons: (1) more U was found and (2) growth of nuclear energy has been slower than at first predicted.

Currently, once-through cycles are in widespread use and the GIF-selected Gen-IV concepts include among them several once-through fuel cycles that may operate for very long time periods, these are: (1) systems based on assumed availability of large quantities of natural uranium and on development of very high discharge burn up fuels and reactor designs and (2) proposed (but not selected) once-through breeder fuel cycles that use natural uranium feed and breed transuranics in situ to harvest 25% of the energy value of the uranium. There have been only limited investigations of waste management aspects of such long-term (century or more) once-through fuel cycles.

Recycle fuel cycles—The original rationale for recycle was to efficiently use actinide resources so as to extend (perceived) limited energy resources. However, the years have taught lessons; now for Gen-IV there are broader goals for recycle: (1) fissile material recovery for reuse, but also (2) destruction of long-lived radionuclides to reduce waste management requirements, and (3) destruction of fissile materials to reduce long-term proliferation risks. Ongoing evaluations of alternative Gen-IV fuel cycle concepts during the Viability R&D phase should consider the implications of these current broader Gen-IV goals. For example, recycle will be deployed on a large scale if nuclear energy is used on a very large scale (which was the case evaluated using the WEC-IIASA cases B and C2 demand growth projections for the FCCG scenario evaluations of potential Gen-IV futures). The size of the global

enterprise has significant implications on how waste management technology should be implemented and what the options are. For example, for large-scale recycle, radioactive noble gas fission product sequesterization may have to replace their free release, which is the current practice; also for large-scale deployment of nuclear, decay heat management will become a major optimization goal for extending repository capacity to limit the number of repositories required.

Mixed and symbiotic fuel cycles—In a small nuclear enterprise, a limited number of plant types have dominated the market because development and engineering costs prohibit deployment of diverse reactor types catering to client preferences or filling differing functional needs. In a large Gen-IV system, however, multiple reactor types tailored to each of the multiple missions can be economically practical and the GIF concept selections acknowledge different missions for different concepts. For example, if long-lived fission products and actinides in spent fuel from reactors that have been economically optimized for electricity and other energy product generation are to be destroyed by transmutation rather than consigned to waste, then reactors specifically designed for the transmutation application are justified. If the system is sufficiently large, the costs and risks associated with a symbiotic mix of specialty reactors are likely to be significantly less than for a system with one or two primary reactors designed for other purposes but modified for actinide burning. (An example of this alternative philosophy is the system proposed by the Kurchatov Institute: thermal reactors for energy production, fast reactors for energy and fissile material production, and molten salt reactors for energy production and destruction of americium, curium, and minor actinides).

In view of the 40–60-year life of power plants, mixed fuel cycles are obviously unavoidable—at least during a transition from current conditions. In such mixed cycle systems, the fuel cycle will function as the mass flow infrastructure backdrop—available to optimally integrate the several reactor types through the medium of mass flows; just as money flows create optimized (efficient) markets—so too can mass flows facilitate optimized nuclear economies with waste management fully integrated.

The innovative options for optimization of integrated waste management during the viability-phase R&D are described next.

3.1.1.2 *Options for Integrated Waste Management.* For Gen-IV, all three of the fuel cycle options (once-through, recycle, and mixed/symbiotic) described above exist among the GIF-downselected concept sets. The traditional structure of the current once-through cycle is not confining for the future Gen-IV cycles; so, the associated waste management systems can be designed from scratch to explicitly allow for alternative future fuel cycles. In all cases, they should be tied together in an optimal manner via mass flows through a system-wide integrated fuel cycle feeding into a system-wide integrated waste management system.

The overall system integration can invoke numerous strategies and new innovations such as:

- Interim storage (above and/or below ground)

- Decay heat management

- Customized waste forms

- Alternate fuels

- Harvesting byproduct fission products

and

20

- Application of breakthrough technologies (should they occur)

- Extended resource base

- Extended disposal options.

Interim storage prior to recycle or prior to repository—Radionuclides decay with time. This characteristic has not been exploited in fuel-cycle system design until now. If one waits for several decades, spent nuclear fuel processing costs and risks drop dramatically because the spent nuclear fuel properties change. (The gamma radiation field is drastically reduced with decay of ^{137}Cs and the decay heat load is down with decay of cesium, strontium, and shorter-lived actinides. The radioactive ^{85}Kr ($t_{1/2}$ = 10.73 yrs) has already decayed.

Time delay can also be exploited for waste disposition by storing high-level waste before geologic disposal to allow time for decay of high-heat radionuclides in high-level waste. Such options may significantly reduce transport and disposal costs for wastes.

Interim repository storage—In all proposed repositories, spent nuclear fuel waste packages are designed to remain intact for many centuries. Repositories are storage facilities; the fissile material does not disappear. Relatively small changes in the repository designs (self-shielded packages, ventilation options) could allow for easy spent nuclear fuel recovery for decades or centuries into the future. Repositories are an appropriate storage place for spent nuclear fuel because the spent nuclear fuel is protected against severe events and disposal remains an option to be decided at any time. Such a retrievable repository strategy would allow societies to walk away with limited future liabilities if other energy sources (fusion, solar) become developed; or it simultaneously maintains future options for recycle as well.

Customized waste forms—Given recycle, different disposal technologies can be developed for different wastes. The wastes can be partitioned and separately disposed with each disposal facility designed explicitly for specific waste types. For example, very low cost methods for separate disposal of high-heat radionuclide wastes (cesium and strontium) have been proposed.

Decay heat management—The heat load of its contents limits repository capacity; however, these limits are not well understood. The capacity limits are probably much higher than current allowable heat loads, which by definition have been designed to be highly conservative. If the repository is ventilated for some time, the capacity increases. In addition, if the spent nuclear fuel is in an easily recoverable shielded package, one can adopt a repository optimization based on more realistic estimates on the allowable repository heat loads, because time is provided for in situ measurements to achieve a better knowledge of actual repository performance. (Some fraction of the spent nuclear fuel could be moved to another future repository if the actual capacity of the repository does not match the initial predictions of thermal response of the repository. In the interim, the repository acts as a highly secure storage facility.)

Another option is to manage the high heat load (Cs, Sr) wastes separately in coolable interim storage and place them into the geologic repository only after a century or so.

Multipurpose casks—Once-through waste management systems such as multipurpose casks that are used for storage, transport, and disposal have only recently been investigated as a means to reduce costs and worker dose by avoiding multiple transfers.

Fuel options—Spent nuclear fuel processing is being considered to allow actinide burning and thus reduce long-term waste toxicity. The higher actinides are primarily generated from the use of a

uranium-plutonium fuel cycle. The quantities of actinides in spent nuclear fuel can be reduced by one or more orders-of-magnitude if a thorium-uranium fuel cycle is adopted rather than a uranium-plutonium fuel cycle. Consequently, any long-term system for burning actinides should be evaluated relative to adoption of a comparable thorium-uranium fuel cycle holding the potential to reduce actinide burning by not generating as many transuranics in the first place.

Harvesting by-products—Given recycle, the potential exists to harvest fission products having commercial value. Examples include rare noble metals, tracer isotopes, and medical isotopes. The revenue generation from byproduct sales would have to offset the extra expense to be a viable strategy – but in some cases might significantly exceed that extra expense.

Application of breakthrough technologies—System studies traditionally evaluate existing technologies and make assumptions about incremental technological progress. Viability-phase system evaluations of integrated waste management should also consider the potential for breaks in the incremental evolution of technology (e.g., radical developments in parts of the fuel cycle could dramatically alter our visions of the future and could initiate paradigm shifts). Two examples are provided:

Deep seabed disposal could conceivably become the lowest cost and safest technology for waste disposal. Such a disposal option would be a paradigm shift to a technology that (1) is international, (2) has unique environmental advantages, (3) has radically different non-proliferation characteristics, and (4) has no capacity limits.

In the 1970s, the disposal of waste in deep boreholes under the ocean was evaluated and the analysis indicated that the potential isolation was superior to traditional geological disposal on continents because it had three added features in addition to geological disposal: (1) enhanced isolation from man, (2) spreading the radionuclides over multiple boreholes in a larger area and thus minimizing the consequences of catastrophic events (volcano, etc.), and (3) ocean dilution. The option failed because of political opposition, limits on the technology, and economics. Since then, however, the oil industry has developed the commercial technology to drill in deep water. There are oil and gas wells in over 1,500 m of water. The technology and economic barriers are beginning to collapse.

There have been proposals for concepts that harvest 25% of the energy value of the uranium in once-through breeder fuel cycles. The feed material is natural uranium. Such fuel cycles would require very-hard-spectrum fast reactors and cladding material of currently unheard of resistance to neutron fluence. Given that a new fluence tolerant clad were to become available, such concepts could provide a radical simplification of the entire fuel cycle, provide significant resource extension, and use one final waste form (the spent nuclear fuel).

3.1.1.3 Recommended R&D. During Gen-IV Viability research and development on (1) integrated mixed/symbiotic fuel cycles and (2) deployed at large scale, it will be important to avoid being locked into the traditional geologic repository strategy but rather to seek to exploit the opportunity to bring innovation to future waste management technologies:

- Exploiting the opportunities that recycle provides for customized waste forms

- Managing decay heat in ways that increase capacity of a given geologic repository

- Exploring the opportunities that breakthrough technologies might offer

- Generally optimizing the entire Gen-IV fuel cycle including waste disposal as an overall integrated system.

The sections below give more technical detail on specific back end technologies, which should receive R&D attention within this systems context. The topics are presented first for issues for an open fuel cycle and second for issues relevant to a recycle-based fuel cycle.

3.1.2　Once-Through Open Future Integrated Fuel Cycle

3.1.2.1　Viability Issues. In the early years of the nuclear power industry, it was thought that uranium was a very scarce resource. The reactors and fuel cycle were built with the assumption that all spent nuclear fuel (SNF) would be processed for recovery of plutonium and uranium. Early repository designers assumed that only high-level waste would be disposed of. As more uranium was found, many countries adopted a once-through fuel cycle. A system designed for recycle evolved into a once-through fuel cycle. As a consequence of this history, there are separate systems for storage, transport, and disposal of SNF. The SNF is transferred from the reactor to a storage system. The SNF is transferred from the storage system to a transport cask and transported to a disposal site. At the disposal site, the SNF will again be transferred into a disposal cask.

If one designed a once-through fuel cycle from scratch, it would likely be considerably different than the current system that evolved over time. Spent nuclear fuel handling would be reduced. Large inventories of SNF would not be stored at reactors in pools. Some of the desired characteristics of such an optimized once-through fuel cycle would probably include the following:

- *Handling*. Spent nuclear fuel handling would be minimized to reduce costs and risks, improve security, and improve safeguards.

- *Pool Storage*. Spent nuclear fuel storage in reactor pools would be minimized to enhance physical security, reduce capital costs for SNF storage in future reactors, and minimize operations. Large-scale storage of SNF in reactor pools implies that when a reactor is shut down, a significant operating and security staff must be retained to ensure safe storage of the SNF. Reactors and waste management operations have different time scales. Large-scale storage of SNF at reactors couples these two systems with different goals and different constraints.

- *Repository Placement*. The safest location for SNF is in a geological repository; thus, there are incentives for enabling repositories to accept SNF at an early time.

- *Repository Goals*. The future is uncertain. This supports the concept of an "Open Future" repository where the repository is designed to (1) ensure safe disposal of SNF, (2) minimize the required financial and other resource commitments by future generations to ensure safe SNF disposal, and (3) allow future generations the option to recover the SNF. Societies may change their preference for fuel cycles because of changes in technology (lower cost reprocessing and recycle technologies), politics, or economics (higher uranium costs).

Recent technical developments suggest that once-through systems with these characteristics may be possible and more economic than the current systems that evolved from systems designed to recycle SNF. Viability research is needed to determine if such systems are practical.

An example of one such system is a multipurpose, self-shielded cask that is loaded at the reactor with SNF and is never reopened. The cask is used for storage, transport, and disposal. It is a sealed, modular, transportable vault for physical protection and safeguards. Overpacks are used to address conflicting requirements between storage of short-cooled SNF and disposal. The repository is modified to allow early placement of SNF. Such systems may be particularly attractive if international SNF storage or disposal systems are developed where safeguards and security issues with each batch of SNF become

transnational. This system allows for alternative futures because the SNF is in a large, shielded, containerized package. Spent nuclear fuel can be stored for long periods of time. The shielded cask characteristics simplify recovery of the SNF from a repository, if desired.

3.1.2.2 Current State of Knowledge. There has been limited work on once-through, open-future integrated fuel cycles. One set of options for such a fuel cycle has been identified. The development of separate storage, transport, disposal, and combined storage/disposal SNF casks provides the basis for a full Multipurpose Cask System that could be used as a once-through, open-future integrated fuel cycle. Figure 1 shows one possible conceptual design of a multipurpose cask system.

Multipurpose casks would be loaded with SNF at the reactor or initial interim storage location. The multipurpose casks would be used for interim storage, long-term storage, transport, and final geological disposal with no additional handling of the actual SNF. The proposed system contains a multipurpose cask, storage overpack, transportation overpack, and repository overpack. The overpacks address conflicting storage and disposal requirements that cannot be easily met by the multipurpose cask.
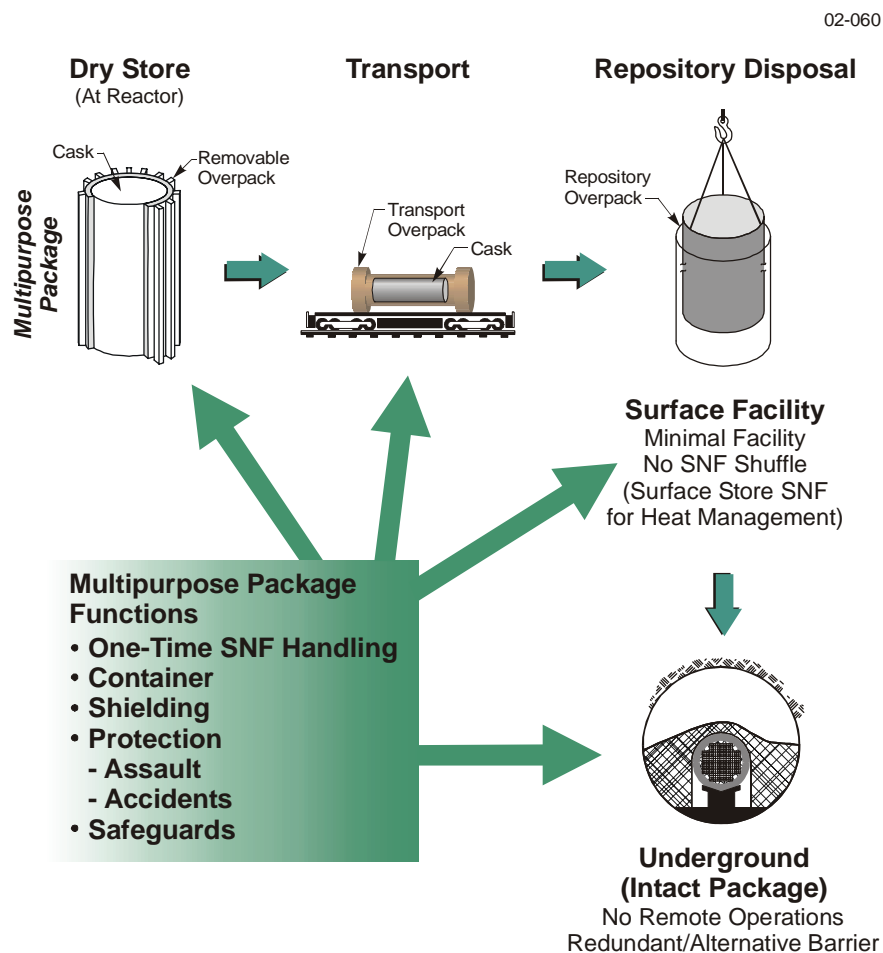
Figure 1. Once-through open-future integrated fuel cycle using a Multipurpose Cask System.

3.1.2.2.1   Multipurpose Cask—The multipurpose cask serves several functions (Forsberg and Haire 2002): (1) handling package for SNF from the reactor to the repository; (2) primary radiation shielding for the SNF; (3) protection against assault, theft, and accidents; and (4) sealed safeguards package.

Shielding—The cask provides all radiation shielding required after some limited period of storage (see storage overpack).

Handling—The multipurpose cask system eliminates most SNF assembly handling (storage to transport cask, transport cask to repository WP, etc.), operations, and associated costs. Except for the initial SNF loading operations, all operations are contact-handled operations—no remote operations. Where protection against aircraft accidents (Germany, etc.) or terrorist assault is required, the cask provides the barrier and avoids the need for handling operations within aircraft-resistant or assault-resistant containment structures. At the repository, the shielded package is an enabling technology if the SNF must ever be recovered from the repository.

Assault, theft, and accident protection—Multipurpose casks have high resistance to assault, theft, and accidents because of their intrinsic characteristics.

1.   The inventory of each cask is limited. In a severe accident or terrorist event, this limits the consequences of any single event, which may offer significant advantages.

2.   Because of the limited inventory of SNF and thus the limited decay heat load, casks are cooled using passive cooling techniques.

3.   The cask provides protection against accidents involving fire such as associated with aircraft collisions. The high thermal inertia of a cask provides significant protection against very high external heat fluxes experienced over a finite (or short) period of time. In storage, sloping of the land allows drainage of aircraft fuel from near the cask and thus limits the fire duration. Alternatively, a foundation of crushed rock allows fuel to flow into the crushed rock where the lack of oxygen extinguishes the fire.

4.   The thick walls of a cask provide significant protection against aircraft impacts and assault. In Germany, metal casks have been successfully tested against aircraft collisions by firing 1-tonne heavy metal poles (the size of jet engine rotors the strongest and most damaging component in an aircraft) at 300 m/s into SNF casks (Droste, Quercetti, and Gololin 2001). Casks have a special property that buildings do not share. Casks do not have foundations; they will move under high impact loadings. This process dissipates the energy in severe events and makes it more difficult to destroy a cask than a building of the same wall structure and thickness.

5.   The multipurpose casks minimize the number of times that SNF is handled (avoids separate transfer of SNF from the pool, to a storage cask, to a transport cask, and to a disposal cask).

    a.   Minimizes the potential for diversion and theft.

    b.   Provides strong physical protection compared to many other SNF storage methods.

    c.   Simplifies tracking of the SNF. Last, the cask size and weight (70 to 100 tons) provides a strong barrier to theft or package diversion. These weight limits prohibit transport by helicopter or typical trucks. The difficulty in transport implies no fast movement of the cask.

3.1.2.2.2   Dry Storage Overpack. If a multipurpose cask system is to be used to its maximum advantage, casks should accept relatively short-cooled SNF. However, short-cooled fuel has much higher radiation and decay heat levels than aged fuel. A removable overpack can provide added shielding and cooling while the SNF cools. The overpack would not be required by the time SNF transport is required.

There are conflicting design requirements for cask storage of short-cooled SNF and disposal of SNF. In storage, the primary design constraint is the need to avoid high temperatures that would degrade the SNF. The storage cask requires a high surface area to volume ratio (small casks or fins) to dissipate heat. For disposal, the primary design constraint is ensuring long-term WP integrity (decay heat levels are low). The WP should have a low surface area to volume ratio to minimize (1) interactions between groundwater and the WP and (2) the cost of expensive corrosion-resistant materials used in the repository overpack. This implies a bare cylinder with smooth surfaces. The use of a removable storage overpack with cooling fins during storage is an engineering method to eliminate these conflicting performance requirements.

The overpack minimizes cask size and weight by providing radiation shielding not required for the cooler SNF during transport and disposal. The shielding requirements for short-cooled SNF are high because of two factors: (1) high radiation levels from short-cooled SNF and (2) added requirements for array storage of SNF. In large storage arrays, the sum of the radiation fields from each cask results in significantly greater shielding requirements than required for a single cask during operations.

3.1.2.2.3   Transport Overpack. The transport overpack provides added protection required for transport. The reactor and repository sites have limited access and tight controls. Such conditions do not exist on public roads and railroads. Most of these requirements result directly or indirectly from this difference.

3.1.2.2.4   Disposal Overpack. At the repository, an outer corrosion-resistant overpack is placed over the multipurpose cask. This ensures the long-term integrity of the waste package. In all repositories, there are limits on the decay heat from each waste package. If there is too much decay heat, the SNF will overheat and degrade. In a multipurpose cask system, the casks are stored until there is sufficient time for the cask heat load to decay to the allowable limits for the repository.

3.1.2.2.5   Repository Design. Repositories can be designed to allow acceptance of shorter-cooled SNF and enhanced capabilities for recovery of SNF by future generations. Typical repository designs include the requirement for retrieveability. Acceptance of short-cooled SNF does require modifying the design to allow ventilation to remove decay heat from the repository for short-cooled SNF.

3.1.2.3   R&D Requirements. There has been limited work on once-through open-future integrated fuel cycles. There may be multiple approaches to create such fuel cycles. One approach using multipurpose casks has been identified. Multipurpose casks have not been previously developed because they demand an understanding of storage, transport, and disposal criteria. While storage and transport requirements are well understood, only in recent years has an understanding of disposal requirements occurred. With this final link, the potential exists for this type of once-through open-future integrated fuel cycle. The primary R&D issues are:

- *Definition of goals and system requirements*

- *Development of storage/disposal overpacks and multipurpose casks*

Overpacks have not been seriously investigated to address the differing requirements of storage and disposal. The viability of multipurpose cask systems depends upon the capability to meet different and sometimes conflicting design requirements in large casks that are economically viable with short-cooled SNF. That depends upon the cask and overpack design.

- *Ventilated repository*

  Repositories can be designed to accept shorter-cooled fuel and easy recovery. Studies for other purposes have identified several technical approaches but these have not been examined in the context of a "Once-through Open-Future Integrated Fuel Cycle."

A 5-year study at ~$1 million per year is required to determine viability. This should include multiple nations to determine viability for alternative repository systems and conditions.

3.1.3 Disposition of Volatile Fission Products with Recycle

3.1.3.1 Goals and Stakes. One can easily suppose that decreasing aerial and liquid radioactive discharges will be an important goal for future nuclear fuel cycles deployed in large scale. In addition, if (as proposed in several concepts) the reactor and the fuel reprocessing re-manufacturing facility are located on the same production site, the necessity to significantly reduce liquid discharges will be even more acute, because of the lack of proximity with the sea, which allows the benefit of a very important isotopic dilution.

Volatile compounds involved in gaseous and liquid discharges of today's commercial reprocessing plants (using the PUREX process) are mainly krypton, iodine and ^{14}C, and tritium. Ruthenium, which can be considered a semivolatile compound, remains in solution in the PUREX process conditions.

To deal with both aerial and liquid discharges abatement, strategies for volatile fission product (FP) and activation product (AP) management must choose an immobilization in matrices suitable for interim storage or long-term disposal of all these radioactive compounds. As glass incorporation is generally not relevant for volatile compounds, gaseous FP management can be considered a major issue in future nuclear systems.

Another factor to consider is the possibly larger amount of ^{14}C to be managed, depending on the fuel options for actinide compounds (carbides, nitrides) and inert materials (TiN, SiC).

In short, the objectives of the processes to be developed are:

- To gather each radioactive volatile compound in an unique stream as far as possible

- To achieve high recovery yield

- To guarantee the stability of the matrix as long as necessary according to the half-life period of each compound.

3.1.3.2 State of the Art. The strategies for volatile compounds management in PUREX-based reprocessing plants are the following:

- Tritium is confined in the head-end process by non-tritiated water scrubbing, then routed toward marine discharges

- ^{14}C, evolved in dissolver off-gases, is trapped as carbonate in aqueous solution and then either routed toward marine discharges or encapsulated in a cement matrix

- Krypton, which is chemically inert and presents a very low radiological impact, is not trapped from dissolver off-gases but simply released from a stack specially designed to favor air dilution effect

- Iodine, evolved in dissolver off-gases, is either absorbed in caustic solution and routed toward marine discharges or trapped on special solid filters.

Many studies have been carried out in the past, or are even underway, concerning more advanced options for volatile compounds management in a hydrometallurgical reprocessing plant. The main trapping and conditioning processes are listed below.

- Tritium—separation by distillation of tritiated water or isotopic exchange on catalytic support, conditioning process by hydruration

- ^{14}C —liquid trapping from dissolver off-gases into alkaline solution followed by precipitation with barium or calcium and encapsulation in a cement matrix

- Krypton—Selective recovery from the off-gases by cryogenic distillation, liquid absorption (for example into solvent as freon), solid adsorption, or selective membrane diffusion, followed by conditioning in pressurized containers with or without inner adsorbents, or by encapsulating in a metal matrix by a combination of ion implantation and sputtering, or by encapsulating in molecular sieves

- Iodine—liquid trapping in alkaline solution followed by precipitation as PbI_2 and incorporation in a specific matrix, or solid adsorption on catalytic support impregnated with $AgNO_3$.

Mostly, these studies consider that the volatile FPs evolve at the dissolution step and are separated in the dissolver off-gas treatment. Another way consists of a direct high-temperature treatment of the fuel before dissolution in order to evolve volatile FPs in a much smaller stream, which enhances trapping efficiency. Unfortunately, recovery yield is not as good as expected and other radioactive elements such as Ru contaminate the gases.

3.1.3.3   Possible R&D Routes. Three types of investigations have to be considered, whatever will be the chosen reprocessing technology (hydro or pyrometallurgy).

1. Behavior of the volatile compounds in the head-end steps of the process

    The observed behavior will depend on:

    a. The type of fuel, especially of the inert matrix and of its ability to retain volatile FP (e.g., diffusion of tritium out of the fuel and coating is quite complete for oxide fuels in pins in fast breeder reactors)

    b. The mechanical process that may partly release the volatile compounds

    c. The chemical processes that might lead to non-volatile forms.

    Basic laboratory experiments will be necessary as soon as the choices of fuel and treatment are made and tests will have to be carried out with irradiated fuel elements.

2.   Development of high-efficiency trapping processes

The probable requirement for very high decontamination factors will lead selection of the most efficient processes and certainly to choose a multiplication of cycles of a chosen process or a combination of two processes. Several techniques seem to be possible and will be worth studying in an extensive way, as mainly liquid absorption or solid adsorption.

Experiments will be performed with non-radioactive isotopes, but will have to consider the most probable chemical forms of the gaseous elements. The main criteria will be recovery yield, separation factor from other radioelements, compatibility with conditioning processes, and safety.

3.   Development of specific conditioning

Krypton and tritium, with half-life of respectively about 10 and 12 years, must be suitably conditioned for temporary storage. For ^{14}C and even more ^{129}iodine, their half-lives are several orders of magnitude longer and final disposal would be preferable. Obviously, conditioning methods and materials will not deal with the same requirements.

Concerning krypton gas storage, the main problems will be heat transfer, the corrosion because of rubidium production, and radiological hazards.

Immobilization of tritium in ceramic or metallic matrices might avoid risks of diffusion, but their elaboration usually requires the absence of water. In hydrometallurgy, tritium management is worth thinking about, especially the point in preliminary isotopic separation to reduce the quantity to be stored.

Regarding ^{14}C, precipitation as a carbonate and encapsulation in a cement matrix seem to be greatly better than bitumen or polymer matrices, or than high-temperature conditioning processes. Studies remain on the choice of the carbonate compound and the cement formulation, but the main requirement will be the stability of the cement matrix.

Development of a specific matrix for iodine like iodo-apatite is essential to deal with the very long required stability.

3.1.3.4   Facilities and Equipment. Several types of facilities will be necessary to perform the above experiments:

- Cold laboratories for basic chemical investigations

- Hot laboratories equipped with gloves boxes and radiometers to carry studies on trace elements

- Hot cells for data acquisitions on irradiated fuel elements

- Equipment development facility for pilot tests of the chosen processes and specific equipment.

3.1.4   Cs/Sr Separations/Treatments with Recycle

3.1.4.1   Viability Issues. Two fission products, ^{90}Sr and ^{137}Cs, have major impacts on the processing of SNF, transport of the high-level waste (HLW), and disposal of the HLW. Both isotopes have half-lives of ~30 years. In SNF, ^{137}Cs is the source of ~90% of the gamma radiation several years after discharge from the reactor. This gamma dose controls many of the shielding requirements for storage, transport, and disposal. In HLW solidification systems (glass melters), it is the primary radionuclide in the off-gas system that must be captured and recycled to the solidification system. This has large cost and operational impacts on operation of processing facilities. Strontium-90 is the dominant
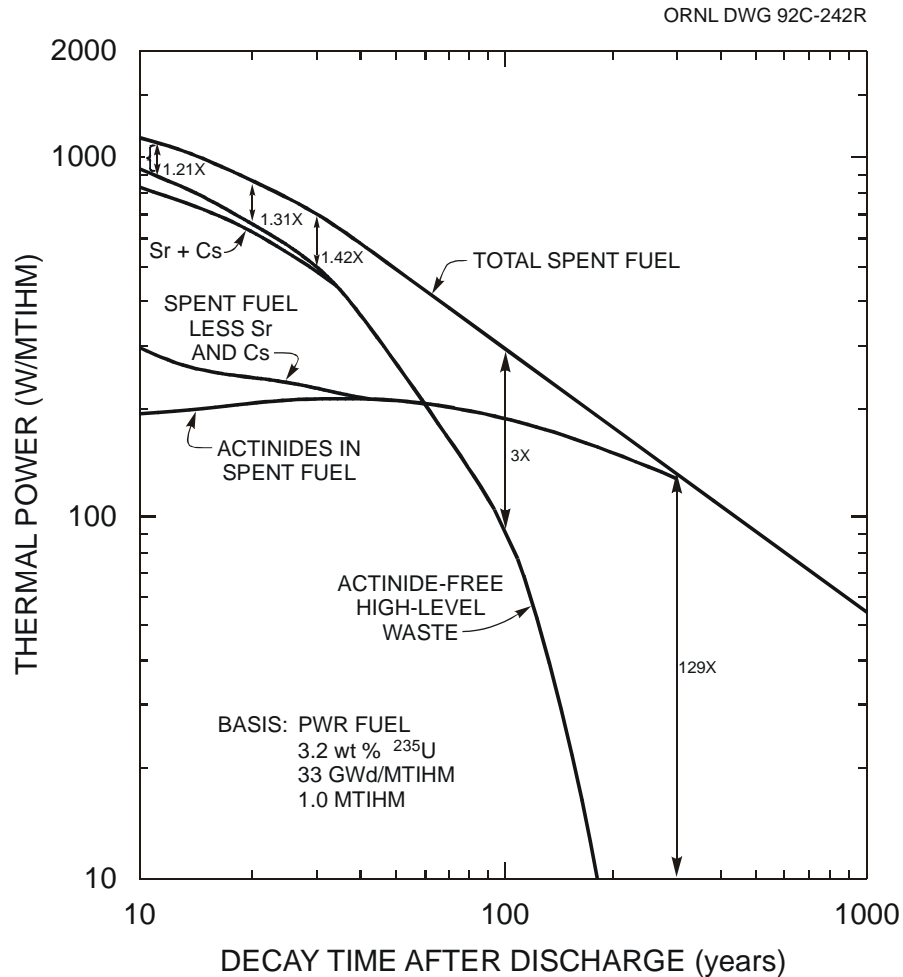
Figure 2. Decay heat from spent nuclear fuel by category.

ingestion hazard for the first few centuries. Cesium and strontium are the primary heat-generating radionuclides (Figure 2) for the first century after discharge of the SNF from the reactor. Repository size and capacity are controlled by the decay heat generation rate in the HLW. If these two radionuclides were not present, SNF processing, HLW transport, and HLW disposal would be greatly simplified, the operational risks would be reduced, and costs would be reduced.

Historically, the assumption has been that all non-fissile radionuclides go to waste in a single waste form. This was based on operational considerations associated with processing SNF. The cost and implications of heat management on the repository were not fully understood. The two isotopes, ^{90}Sr and ^{137}Cs, have half-lives of ~30 years. This is considerably shorter than the long-lived radionuclides in HLW that require geological disposal. The half-life of these radionuclides is sufficiently short that these radionuclides (1) could be stored until most of the radioactivity has decayed away and then disposed of or (2) lower-cost disposal methods could be deployed.

If SNF is processed, there are two alternatives: (1) dispose of the HLW in a repository or (2) separate out and separately manage strontium and cesium. Separation of strontium and cesium involves increased costs in the processing facilities but considerable savings in waste storage, transport, and disposal. The question is which option is economically and environmentally superior?

3.1.4.2   Current State of Knowledge. For a variety of different purposes, ^{90}Sr and ^{137}Cs have been extracted from HLW. These technologies are reasonably understood. However, the implications of separately managing a low-heat HLW and the high-heat ^{90}Sr and ^{137}Cs are not well understood. No detailed studies on separate disposal of these two isotopes have been undertaken. A few limited studies have identified disposal options (see references) and issues.

The benefit of separate management of ^{90}Sr and ^{137}Cs is strongly dependent upon whether the minor actinides are also separated from the wastes and destroyed (actinide partitioning and transmutation). There are only two sets of intermediate and long-lived decay-heat sources in the repository: (1) the fission products ^{90}Sr and ^{137}Cs and (2) the actinides. If both are removed from the HLW stream, there is little decay heat in the repository and radial reduction in repository size and cost may be possible.

3.1.5   R&D Requirements

The preferred waste management options involve a tradeoff between managing ^{90}Sr and ^{137}Cs separately or with other radionuclides. There are two cases: (1) the HLW contains the minor actinides (reprocessing with uranium and plutonium recovery) or (2) HLW does not contain the minor actinides (processing with all actinides removed and destroyed). In this second case, the size of the repository is drastically reduced and the characteristics of the repository are radically altered. Required R&D activities include:

- The options for separation, solidification, storage, transport, and disposal of strontium and cesium must be identified and sufficiently developed to be well understood, examined with respect to performance, and costed.

- The impacts of ^{90}Sr/^{137}Cs removal on SNF processing facilities, HLW solidification, HLW transport, and HLW disposal must be identified and developed sufficiently to be well understood; examined with respect to performance; and costed. Because of the large impact on decay heat within the waste and radiation levels of HLW, the transport and repository systems may be significantly simplified by removal of cesium and strontium.

3.2   Multi-Partitioning Strategies

3.2.1   Introduction

A recycling-based, sustainable society is being sought to serve the expanding global population. The fast reactor (FR), in using its neutron excess to manufacture fissile fuels from abundant fertile isotopes, relies on a recycle-based fuel cycle to attain sustainability. Given that recycle is an essential element of reaching Gen-IV sustainability goals, further benefits may be attainable by means of an extended partitioning of the fission products into ones of value, ones suited for short storage before disposal, and ones suitable for special treatment before disposal.

3.2.2   Fuel Cycle System in Generation IV

Nuclear fuel cycle is outlined from starting with the extraction of U from ore to terminating with the disposal of radioactive wastes according to policies that differ from country to country. The FCCG examined fuel resource inputs and wastes for the range of potential Generation IV fuel cycles such as once through, partial recycle, full (Pu, U-233) recycle and all actinide recycle as shown in Figure 3, consistent with projected energy demand scenarios on a global basis. The common argument point is that the fuel cycle is necessary to sustain the nuclear energy, especially taking notice of effective waste

management. And the symbiotic mixes of thermal and fast reactor concepts will be introduced with features of fuel
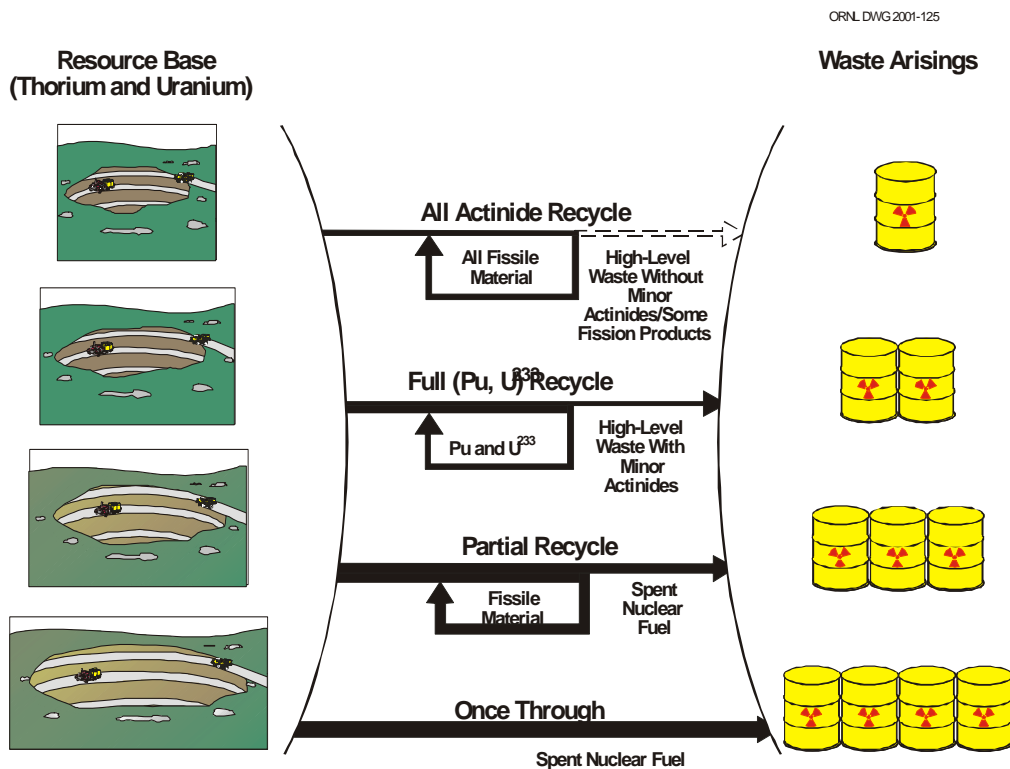
Figure 3. Fuel cycle alternatives.

recycle in the 21st century. Furthermore, the new idea for fuel cycle might be considered to reduce arising high-level radioactive wastes as low as practical because of the limitation of the repository candidates in long-term base scenario of nuclear development sustainability.

3.2.3   New Approach in FR Fuel Cycle System

At first, the volume reduction of HLW is discussed consistent with FR, reprocessing, and fuel manufacturing and is recognized as the key factor in the future fuel cycle. When the practicality of partitioning technology is considered, the elemental separation (rather than isotopic) is basically adopted because it will take a long time for the isotope separation method to reach commercial base. Brainstorming on potentially achievable enhanced partitioning concepts by engineers from each of the diverse fields of the nuclear enterprise—reactor design, reprocessing, fuel manufacturing, and repository––the development motto is defined as "Let's get rid of any impediment elements against the fuel cycle and repository instead of merely extracting the available elements such as U and Pu." This also leads to "from clean fuel and dirty waste to all actinide fuel and clean waste". For example, core design engineers assess incoming elements from the standpoints of neutron absorption. Fuel manufacturing engineers evaluate them based on radiation exposure, heat generation, quality assurance, comparability with clad, and so on. Engineers in the repository rate them in the light of heat generation, radiotoxicity and vitrification process, and so on. Finally, the ideal element-wise partitioning scheme based on all needs for each link in the FR, reprocessing, fuel manufacturing, and repository fuel cycle chain is summarized with the classification of recycle use in FR, resource as noble use, specific treatment for transmutation, storage, low-level or stable waste, vitrification as HLW. This ideal partitioning is shown in Figure 4. Then, the
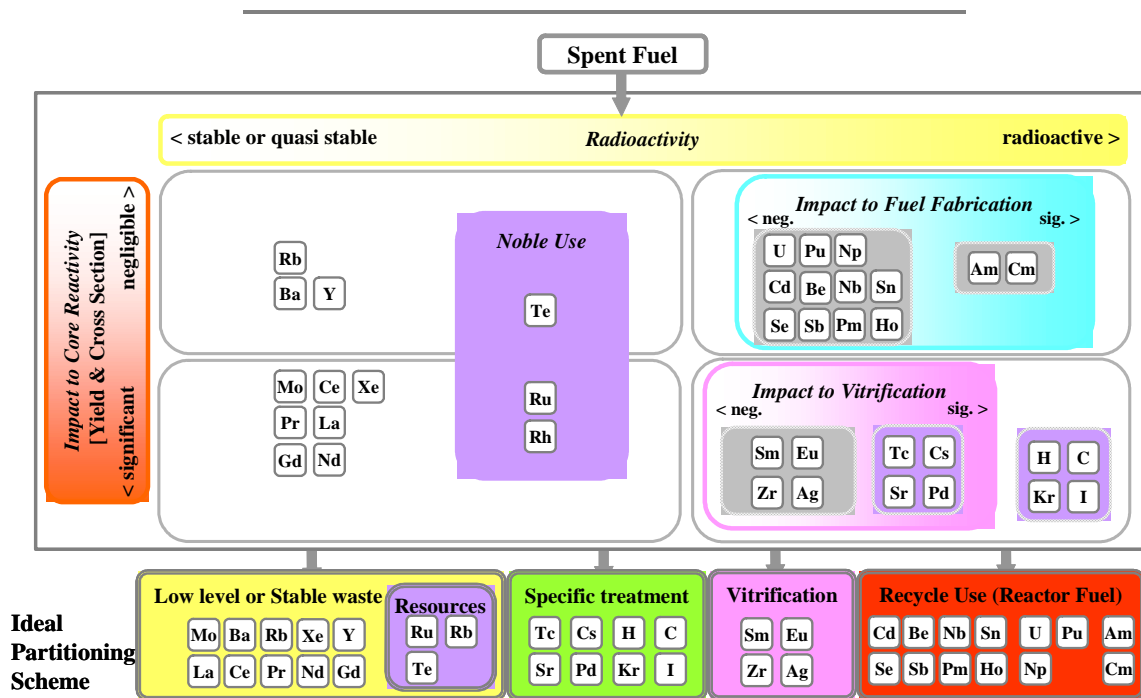
Figure 4. Ideal partitioning by need.
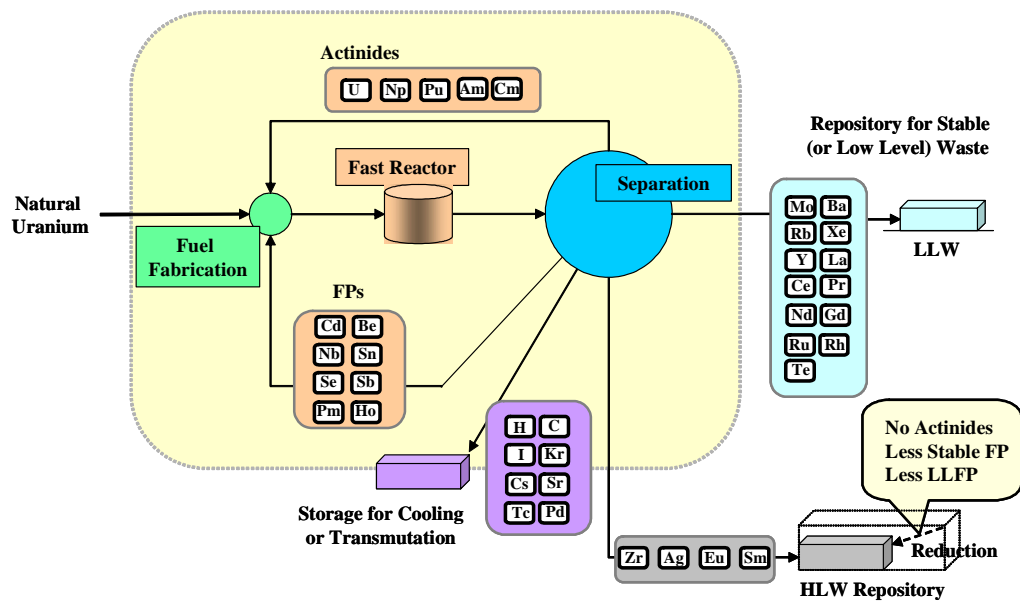
Figure 5. Material flow in "New Concept."

material flow in the new concept is described, as shown in Figure 5 with the endorsement of neutron feasibility analysis of the multiple recycling.

As the next step, introducing innovative technologies embodies actual processes for aqueous and non-aqueous reprocessing. Figure 6 shows the aqueous partitioning with the following technologies:

- Voloxidation—removal of volatile fission product elements by high-temperature oxidation or reduction.

- NCB precipitation—separation of U, Np, and Pu oxidized by ozone, then Am, where possible for oxidization by electrolysis.

- Catalytic electrolysis extraction—separation into three parts as rare elements (Mo, Zr, and others).

- Thin film separation—separation into two parts of remained FPs using DC18C6 thin film

- Ion exchange purification of each element. Non-aqueous partitioning is also shown in Figure 7.

The HLW generated per unit energy based on the waste mass analysis of the aforementioned multi-partitioning new concept is reduced by about one order of magnitude compared to the conventional recycle procedure. Since the new concept has the very attractive point, the advanced aqueous method, oxide electrowinning method and metal electrorefining method that the new concept is applied to are being investigated by in-depth design study and experimental tests of key technologies under the domestic and oversea cooperation studies in the second phase of the feasibility study on commercialized FBR cycle system in Japan.

Figure 6. Aqueous partitioning.

Decladding Bond Na Removing

Rare Gas : Xe, Kr, He
Volatility FPs :
Se, Br, Rb, Cd, Te, I, Cs → Partitioning by Vapor Pressure

Cd Distillation

Hulls, Residue :
Se, Zr, Nb, Mo, Tc,
Ru, Rh, Pd, Ag, Cd,
In, Sn, Sb, Te

Electro-refining

Solid Cathode : U, Zr

Liquid Cd Cathode : Np, Pu, Am, Cm

Tc, Pd Partitioning

Li-K Extraction

Np, Am, Cm

Pd   Other FPs

Rare Earth :
La   Pm, Gd   Yb

Electro-refining

Low Level Waste
Rare Earth, Mo etc.

Treatment of Spent Salt
=> Bonded Zeolite
(Salt-Occluded)

Electro-reduction

Halogen : Br, I → Separation by melting point (Subzero Treatment)

Zeolite Filtering (Cs)

Zeolite Column : Cs

Zeolite Column : Other FPs

Zeolite Filtering (Other FPs)

Interim Storage
Cs (, Sr)

High Level Waste
Other FPs

Figure 7. Non-aqueous partitioning.

3.2.3.1 ***Discussion.*** The ideal partitioning approach is introduced and some examples applied to aqueous and non-aqueous reprocessing method are shown. The seed technologies will be developed in the world and we hope that they will improve ideal partitioning approach. Of course, we must bear in mind pursuing not only soundness of the technology but also the economic viability of the new concept from the viewpoints of the commercialized FR fuel cycle system and the reduction in secondary arising waste.

3.2.4   P/T Thorium Cycle

3.2.4.1 ***Thorium Fuel Cycles.*** What issue of crucial importance must be resolved during the Gen-4 viability R&D phase?

If Th-fueled power reactor should be considered again as a possible option for future energy supply, a significant research and development effort on the Th fuel cycle will be needed. Main thrust of the research program would be directed to:

1.   The examination of various fuel cycles for the optimal way to introduce Th into the current generation of reactors without upsetting the ongoing power program and development of an appropriate reprocessing technology

2.   An economic fabrication technique for the Th-based mixed oxide fuel (e.g., ThO_2/PuO_2) similar to the fabrication of the U-based MOX (UO_2/PuO_2).

Some of the reasons why Th should be considered in preparing the long-term nuclear future are given below by decreasing order of priorities and increasing order of importance:

- Contributing to burning of excess Pu from energy production (LWRs) and from weapons dismantling.

- Generating less long-lived waste—when a Th fuel cycle is used, much less transuranic elements are produced, compared with U fuel cycles.

- Going to higher burn-ups—U-233 bred from Th is for all neutron energies, either thermal ones or epithermal, or fast ones—the best of the three nuclear fuels, U-235, Pu-239, U-233. This means that U-233 will be a good fuel in any reactor type.

- Going to high temperatures—U-233 keeps its good neutronic properties also here, much better than either U-235 or Pu-239. These properties and the good mechanical properties of Th fuels have led naturally to recommend the Th cycle for high-tem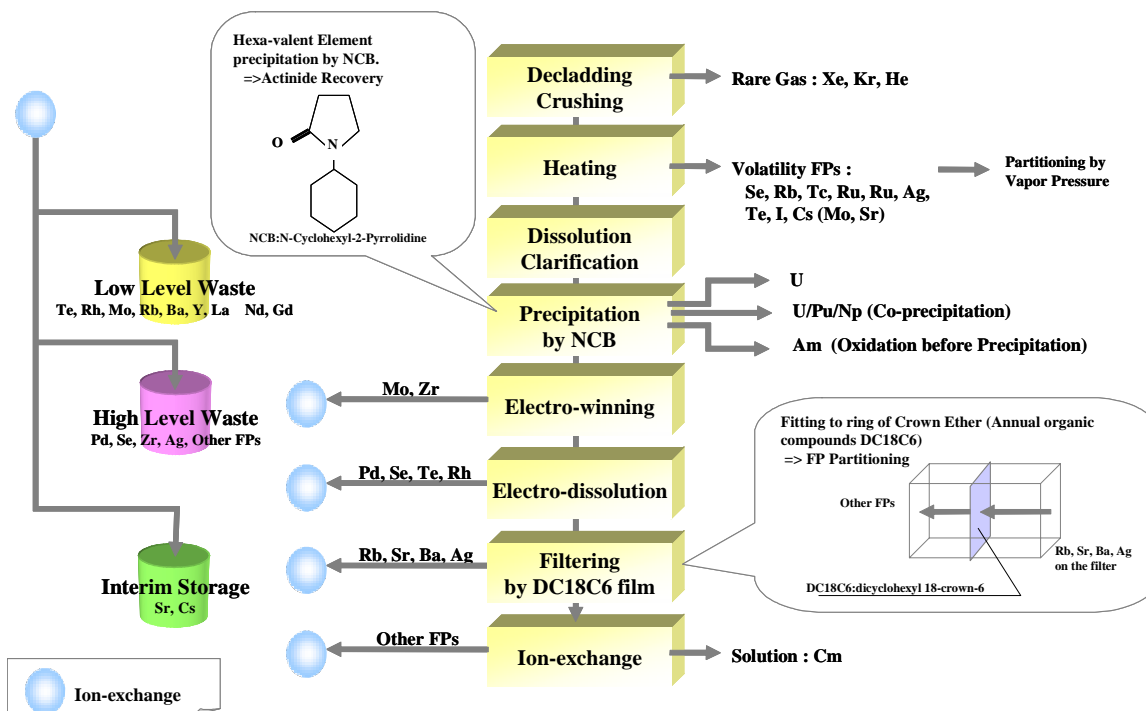perature reactors. High temperatures are sought to (a) use conventional turbo-generating plants, (b) obtain good thermodynamic yields, and (c) aim at chemical syntheses like ammonia, hydrogen, etc.

- Sustainability, in view of limited U reserves—Th (around 6 ppm in soil) is about three times more abundant in nature than U.

What is known currently, what is still not known?

Starting around 1955–1960, many industrial prototypes based on Th were built in parallel with those based on U, involving main industrial companies with the scientific knowledge of the research laboratories, for example, Oak Ridge National Laboratory in the U.S. General Atomics, Babcock and Wilcox, Allis Chalmers, Westinghouse, and others in the U.S. have taken an active role in this industrial development. Germany, the UK, France, and Russia have followed suit. Th extraction plants are installed in many countries—such as the U.S., Germany, and France—and have each separated in the past about 2,000 tons of thorium, part of which is still available. Today, Japan and China continue to develop the Th fuel cycle technology—the case of India being quite special.

The U resources (50,000 t) in India, where energy self-independence for 1 billion inhabitants has to come from nuclear energy principally, are not sufficient even with the aid of breeder reactors. However, the large Th deposits (360,000 t) will permit a durable development of nuclear energy in this country.

3.2.4.2   Emerging Advanced Reactor Concepts. In contrast to the ^{238}U/^{239}Pu cycle where breeding can be obtained only with fast spectra, the ^{232}Th/^{233}U cycle can operate with either fast or epithermal spectra. As a consequence, Th fuel cycle reactors can use, with suitable reprocessing, either solid fuel with fast neutrons or molten salt with moderated neutrons.

Several new reactor concepts based on Th fuel cycles are under consideration at present; those include:

- Light water reactors with fuel based on Pu oxide (PuO_2), Th oxide (ThO_2) and/or U oxide (UO_2) particles arranged in fuel rods.

- High-temperature gas-cooled reactors (HTGRs) where the fuel elements used up to now are of two types: (1) hexagonal vertical graphite fuel elements or (2) graphite balls of 60-mm diameter that flow slowly down the reactor core, so-called pebble bed reactors. An example is the modular reactor Pebble Bed Modular Reactor (PBMR) conceived in South Africa and now being developed by a multinational consortium.

- Molten salt reactors, already studied in depth in the 1960s by the U.S. (MSBR), are now being revived because of the availability of advanced technology for the materials and components. Furthermore, the better breeding and radiotoxicity performance of the MSR are mainly due to

much lower inventories and a greater freedom in fuel reprocessing. Therefore, the already considerable knowledge of molten salt systems should be at the rate of 12 tons per year for ^{232}Th/Pu ^{233}U completed by new simulations and experiments, particularly on safety and chemistry aspects.

- The Advanced Heavy Water Reactor (AHWR) was developed in India. In Canada, the CANDU-NG with a 250-MWe design is light water cooled. The main part of the core is subcritical with Th/U-233 oxide mixed so that the system is self-sustaining in ^{233}U. A few seed regions with conventional MOX fuel will drive the reaction and give a negative void coefficient overall.

- Pu disposition—Today, MOX (U, Pu) fuels are used in some conventional reactors, with Pu-239 providing the main fissile ingredient. An alternative is to use Th/Pu fuel, with Pu being consumed and fissile U-233 bred. The remaining U-233 after separation could be used in a Th/U fuel cycle. Consideration can also be given to deploying fertile-free (inert matrix) Pu fuels in current-day reactors.

- Accelerator Driven Systems (ADSs)—again, the use of Th instead of U means that less actinides are produced in the actual reactor.

In the frame of the 5th European Framework Program, a project called **Thorium Cycle** has been launched to investigate the irradiation behavior of Th/Pu fuel at high burnup and to perform full core calculations for Th-based fuel with a view to supplying key data related to Pu and minor actinide burning. Two irradiation experiments will be carried out: (1) four targets of oxide fuel (Th/Pu, U/Pu, U and Th) will be fabricated, irradiated in the HFR in Petten; (2) one Th/Pu oxide target will be also irradiated in KWO Obrigheim. All irradiated materials will of course be submitted to detailed characterization after irradiation

3.2.4.3 ***Head-End and Back End Processes.*** Comparing the Th to the U mining, the following can be said:

- From the so-called mining residues, Rn-220 is released in the case of Th (Rn-222 in the case of U). This means, in general, a smaller air concentration, but a larger impact by inhalation.

- Monazites, the most likely ores to be exploited first, in case of an extensive use of Th as nuclear fuel. However, the Th concentrations in monazite are very large, such that the total radioactive waste mass produced from mining is about two orders of magnitude lower than for U. Furthermore, costs can be shared with rare earth mining also contained in monazites.

- Long-term real radiological impacts due to U daughters is also two times lower compared to U because of the short half-life of Ra-228 (5.7 years) as compared to Th-230 (77,000 years) and Ra-226 (1,600 years) for U.

- According to recent ICRP-72, inhalation and ingestion dose commitments are about 5, respectively, 2 times higher for Th. Thus, more rigorous environmental monitoring for workers and the public will be required.

Regarding the radiotoxicity of waste arising from Th-based fuel, it is evident that the closed cycle gives the best performance in comparison to U-based fuels especially if fast reactors are used.

What R&D needs to be undertaken?

In a Th-fueled reactor, Th-232 by neutron absorption produces first Th-233 and Pa-233, which has a 27-day half-life, produces U-233:

Th-232 + n ----> Th-233 -(22 m)--> Pa-233 -(27 d)--> U-233 (1.5×10^6y)
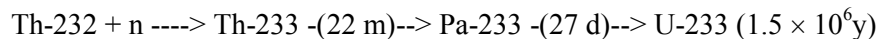
It appears from what precedes that in a sustainable Th fuel cycle sooner or later U-233 formed should be separated for incorporation in efficient nuclear fuel elements. This means that reprocessing is an integral part of a sustainable Th fuel cycle.

The rather long half-life of Pa-233 results in a reactivity surge after reactor shutdown due to U-233 production, and this must be taken into account for the reprocessing of this fuel type.

One of the principal drawbacks of the Th cycle today is the presence of hard gamma emitters (2 to 2.6 MeV) among the descendents of the Th-232 isotope, and especially of U-232, an alpha emitter of 72 years half-life. This U isotope is always present along with U-233 at concentrations ranging from some tenths to some hundreds ppm. Many experts in the world view this characteristic also as an important positive feature, because it makes the Th cycle resistant to proliferation. However, it obliges the manufacture of Th-based fuels completely in a remote, gamma-shielded environment—a very expensive technique that only starts to be mastered with the fabrication of UO_2-PuO_2 MOX fuel elements and would need further adaptation for a Th fuel cycle. Schemes for laser isotopic cleanup of U-233 are under investigation at present and could, if desirable, simplify the fabrication process.

Another problem to be solved is the fuel fabrication for HTGR fuel. To minimize the release of volatiles during irradiation at high temperatures, the fuel generally consists of small kernels (about 1 mm diameter) of enriched U/Th carbide coated with tight pyrolitic carbon layers (BISO type) or with a silicon carbide overcoat (TRISO type). These particles are embedded into amorphous graphite into what is called "compacts". These expensive and lengthy fabrication processes have to be streamlined and remotized for the above-mentioned reasons.

A much simpler and more favorable case regarding fuel fabrication is molten salt reactors, because no fabrication is needed.

Concerning reprocessing, the head-end process (i.e., the fuel dissolution) is not as easy as for UO_2; the problem can, however, be overcome with a buffered fluoride addition in the dissolver solution. To muster such a process at industrial scale, a large development program will be required. In principle, the so-called THOREX process developed in the U.S. and in Germany for reprocessing of Th fuels is similar to the PUREX process. However, direct utilization of existing industrial facilities like the ones in La Hague or Sellafield is not possible due to the bad extraction behavior of Th; therefore, a salting out agent is required. In addition, the separation from U is not possible as for Pu with a simple valency change; it can only be achieved through the difference in distribution factors between aqueous and organic phase.

The problem of the non-extractable decay product, Pa-231 (3.2 104 years half-life), also needs to be solved; a selective Pa adsorption, as proposed in the THOREX scheme, needs to be further developed.

The particle fuels, which have multiple carbide coatings as in the case of HTGR, are not easily reprocessed and might need a dedicated process to physically or chemically separate the fuel kernel from the coating. The oxide kernel could then be treated as a conventional oxide or carbide fuel.

Regarding the alternative reprocessing route (i.e., pyroreprocessing), almost nothing is known concerning the handling of Th-based materials, perhaps with the exception of molten salt reactors, where some research was carried out already in the early 1960s. For solid Th fuel reactors, however, molten salt technologies for reprocessing require a completely new R&D program. A number of problems specifically related to the Th fuel cycle need to be solved including:

- Fuel dissolution in molten salt, especially in the case of coated particles

- Conversion processes (e.g., ThO_2 or ThC) if metallic concepts have to be applied

- ^{232}U behavior also in view of subsequent fuel fabrication

- ^{231}Pa problem—can possibly be transferred to the waste treatment part of the process

- Fuel fabrication after reprocessing.

For all these processes, a detailed database of thermodynamic and electrochemical data in various molten salt systems is to be established.

These largely experimental R&D programs rely, of course, on the availability of corresponding facilities:

- Fuel fabrication (remote) facilities

- Equipment to determine basic thermo-physical properties

- Fuel irradiation reactors (also fast neutron spectrum), including test reactors for safety transient testing

- Hot cells and equipment for post-irradiation examination

- Shielded caves for (remote) recycle experiments in the case of pyroreprocessing

- Adaptation of existing facilities in the case of aqueous reprocessing.

What are the costs involved?

Cost numbers can be only very indicative because, as shown above, many reactor types are involved and also the choice of a possible reprocessing technology is not made. The new fuel types will, in any case, require the development of fuel fabrication and recycle technologies and their demonstration in prototype facilities of appropriate size and plant throughput rate. Technology development could run for 5 to 8 years. Indicative costs in millions of dollars are given below.

Base technology	100–300M
Fuel fabrication and irradiation	200–500M
Recycle technology (aqueous and pyro)	400–700M
Prototype demonstration	200–300M

3.3   Concept-Specific, Generic Fuel Cycle R&D
for Middle Links of the Fuel Cycle

3.3.1   Generic Issues for Gen-IV Recycle Strategies

The concept-specific fuel cycle relevant R&D needs for the middle links of the fuel cycle chain (fabrication, recycle, re-fabrication) are identified and recommended in the Technical Working Group's R&D scope reports. Examination of those identifications and recommendations shows that several issues are generic and deserving of a crosscutting R&D program.

Recycle-based Gen-IV fuel cycles are proposed for the following GIF-selected concepts sets:

- Supercritical water (fast option)

- Fast gas

- Fast Na or lead alloy

- Molten salt.

In addition, for the first three of these, both aqueous and pyrometallurical recycle options remain open. All recycle options intend to recycle all transuranics for fission consumption, but the strategies differ as to:

- Whether coextraction is done—with subsequent remote refabrication

- Whether separate extractions are done—with remote refabrication for some, but not all, of the recycle products

- Whether separate extractions are done—with prompt reblending of separate products and remote refabrication.

Thus, the broad strategy for execution of recycle/re-fabrication remains an open issue for Gen-IV across the several recycle-based concepts. The tradeoffs in economics, sustainability, and nonproliferation goals—while dependent on fuel type—nonetheless are generic, crosscutting issues. The broad generic issues, which motivate crosscutting recycle R&D, are briefly described below:

- Need for flexibility with respect to fuel type

- High recovery at industrial scale with few-step coextraction processes

- Continuous efforts in all cases to minimize secondary waste production

- Potential for hybrid processes (combining features of aqueous and dry)

- Compactness (for cost control) and for collocation with power plant.

3.3.1.1   Advanced Aqueous Processes—Main Incentives for Crosscutting R&D. Today, the PUREX process operates at industrial scale for the recovery of uranium and plutonium from oxide spent

fuel. The plant operations experience (for instance, the La Hague plant in France) shows that such aqueous processes:

- Achieve very high recovery yields (about 99.9%) and purification yields (average decontamination factors [DFs] higher than 10^6 for both uranium and plutonium)

- Produce low amounts of technological waste.

The Gen-IV goals provide motivation for "advanced aqueous processes" to further improve this industrialized process. Areas for improvement could be the following:

1. First, there will be a need for increased flexibility. The PUREX process has proven to be efficient for oxide fuels (both UOX and MOX); however, future Gen-IV fuels may be different, with possible changes in the nature of actinide compound (from oxide to nitride or carbide) and also in the shape and the nature of "cladding" (e.g., SiC-coated particles). This calls for process adaptation, essentially in the head-end part, to be able to solubilize other kinds of fuels besides homogenous oxide pins. It can involve the core of the process too, with a possible need of more radiation-resistant extractants to cope with high burnup or shorter cooled fuels.

2. Another reason is the strategy of "all actinide recycle" (not only uranium and plutonium, but also minor actinides) adopted in some Gen-IV fuel cycles. Research is already in progress in several countries to add complementary extraction steps to the PUREX process.

   However, a key point in this field is to reach a co-extraction of all transuranics, from spent fuel to recycled products, without intra-group separation. This would be a real improvement, especially considering non-proliferation concerns. Such a goal could be reached by developing new extracting systems able to recover both major and minor actinides in a commixed product.

3. A third important motivation is to increase the compactness of the process, to achieve a more cost-effective recycling operation. This concerns process chemistry—trying to increase the efficiency of elementary operations (and come to a "one single extraction cycle" process, in place of today's multi-cycle operation). However, process technology is concerned as well—there is room for more compact, smaller-sized technologies to achieve the unit operations, either for the dissolution step or for the core of the process (e.g., with the possible development of new extractors technologies).

3.3.1.2   Pyroprocesses—Main Incentives for Crosscutting R&D. Dry pyrochemical processes appear rather attractive for future nuclear fuel cycles. They offer several presumed advantages, the most important among them being the following:

- Ability to process refractory and hot fuels, due to the high solvating power of molten salts and the radiation-resistant feature of the chemical reagents involved (no organic radio-sensitive molecules)

- Compactness (limited number of transformation steps—actinide is early recovered in the form suitable for recycling).

Some other features of the pyroprocess can be mentioned, like suitability for "onsite processing" (i.e., collocation of power plant and recycle facilities as a proliferation-resistant concept [grouped recovery]).

The main goals for the R&D to be achieved in this field are the following:

1. First, to assess the ability of such processes to recover actinide with high efficiency (about 99.9%) from molten salts solutions. This has not yet been demonstrated for all actinides and at a representable scale.

    The R&D could aim at reviewing the diverse process options, either for the media (chlorides, but also fluorides) or for recovery means (electro-deposition, extraction by molten metals, selective precipitation . . .) in order to check how the targeted recovery yields could be obtained.

2. Second, to increase the flexibility of the process (as with the aqueous, a need for adaptation to the different kinds of fuels: oxides, but also "advanced fuels"…).

3. Third, to achieve the conditions of an industrial operation: back-end operations (salt recycling, water conditioning, . . .) and operability regarding the diverse criteria governing industrial operation (technological waste, process control, . . .).

3.3.1.3   *Particle Fuels—Main Incentives for Crosscutting Recycle R&D.* Particle fuels are proposed for the GIF-selected gas reactors—fast gas and VHTR—which are proposed as closed cycle and open cycle, respectively. More broadly, however, particle fuels are intended for two applications: (1) very high temperature service conditions and/or (2) very high discharge burnup. Moreover, investigative work in other (than Gen-IV) DOE and worldwide programs considers them as candidates for inert matrix fuels (i.e., fuels having no ^{238}U or ^{232}Th) in P/T applications. Because of the targeted service conditions, the particle fuels are intentionally robust (and resistant to degradation, including recycle); nonetheless, future systems (and fast gas in particular) seek to recycle particle fuels. A generic need arises to develop technologies for recycle and re-fabrication of particulate fuels.

3.3.1.4   *Secondary Waste Minimization—Main Incentives for R&D.* When recycle is used with a goal to reduce the mass, volume, and long-term radiotoxicity of the HLW, it is imperative that the production of secondary waste from the recycle operations does not negate the gains made on the HLW. Societal impact of waste management is affected by both the source term and the impediments to societal exposure to the source term. The HLW is destined for geological disposal (i.e., in a way that processes powerful impediments to societal exposure), whereas the impediments applied to the secondary waste are substantially less owing to a lesser source.

As new, innovative recycle/re-fabrication technologies are developed for Gen-IV, continuing attention must be paid to avoiding increase in the secondary waste source term (per unit energy). In fact, once deployed, a continual campaign can be expected forever to further reductions in secondary wastes.

3.3.2   Group Separation of Actinides (Aqueous Process)

3.3.2.1   *Goals and Stakes.* In order to reduce the waste impact and to preserve the natural resources of energy—two important aspects of sustainable development—one of the goals of fuel treatment for future nuclear systems is to recover and recycle the actinides. As aqueous processes have in the past been the only mature process for fuel treatment, the objective here is to develop a new process able to answer to the needs of the future nuclear systems: recovery and recycling of all the actinides. Today, different options are available to fulfill this goal, but are based on multiple steps involving intra-group separation. As the objective here is to recycle the so-called "minor actinides"—neptunium, americium, and curium—together with plutonium, the goal is to develop a process able to recover all this actinides together without any intra-group separation.

This will significantly increase the compactness and the economical consistency of the process.

The proliferation-resistant feature of the fuel cycle could also be increased. As there is no separation of plutonium from the other actinides, the strategic value of weapons-usable material is decreased and the presence in significant amounts of americium and curium isotopes greatly increases the spontaneous neutron, heat, and radiation generation rates and then enhances the technological barriers.

Finally, shortening the fuel cycle reduces the actinides hold up. This means short cooling times before fuel treatment—hence the use of extractants with high radiolytic stability. This characteristic is a strong requirement to allowing industrial implementation.

3.3.2.2 **State of the Art.** If the literature offers numerous processes able to separate uranium, plutonium and neptunium, americium and curium from fission products, all these process were developed with an objective of, at least, a separation of uranium and plutonium from the other actinides. For instance, the process developed in France in the frame of a Partitioning and Transmutation strategy is based on the following steps :

1. Recovery of uranium, plutonium, and neptunium separately using the PUREX process

2. Separation of americium and curium from fission products using the DIAMEX and SANEX process

3. Separation of americium from curium.
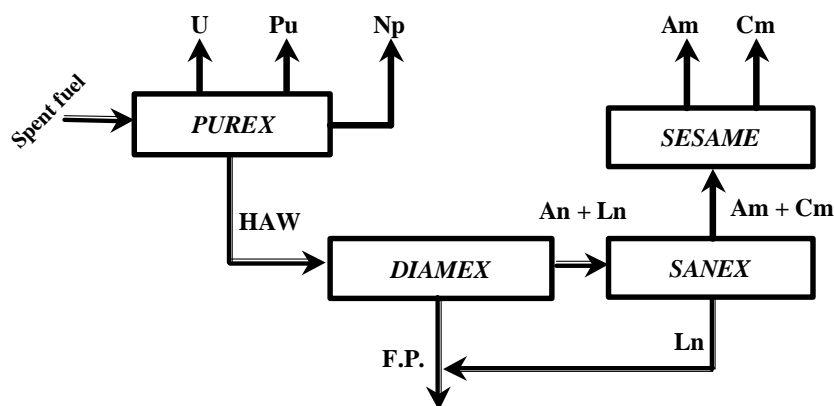
French minor actinides reference partitioning process.

Indeed, there is no process today that is able to separate the actinides all together without any intra-group separation.

3.3.2.3 **Possible R&D Routes.** Taking into account the number of processes available from literature for actinide recovery, but also, at the same time, the absence of processes able to fulfill the criteria of recovery of a transplutonium element without intra-group separation, a four-step development program is proposed:

Selection of concepts—One of the tasks here will be to assess ability of the known process to fulfill the goal with some minor or major adaptation. As an example, the ligands (O-bearing and N-bearing) chosen in the frame of the DIAMEX-SANEX concept should be able to separate plutonium together with americium and curium from fission products. The second task, having a more broad and open approach, will be to investigate the different methods able to answer to the objective. Chemical conditions will be

examined in order to confer to the transplutonium element's similar behavior versus liquid-liquid extraction (aqueous media or organic extractant).

The use of very basic tools, as quantum chemistry, could be useful in the research of these new organic extractants and also to propose possible route to strengthen their radiolytic stability. Experiments, at laboratory scale involving basic research, will be necessary in order to assess the feasibility of these concepts.

Validation of concepts—Having chosen some few promising process, it is necessary to perform tests in more realistic conditions to confirm the feasibility of the concepts; for instance, counter-current tests will be carried on with surrogates for liquid-liquid process notably to better evaluate the practicable performances and the behavior of some fission products. Depending on the progress of the studies concerning the dissolution of the fuels, the compatibility with the studied media will be assessed. The hydrolytic and radiolytic stability of the solvent and the main degradation products will be determined to be able to propose a solvent regeneration treatment.

Thus, a final selection of one or two process will be done considering criteria that are specific to fuel reprocessing, in particular:

- Compound stability

- Efficiency, robustness, and simplicity of the process

- Compatibility with a waste confinement material and process.

Process development—At this point, an optimization of the chosen process will be carried out, taking into account the dissolution media and the re-fabrication constraints. This task will include studies in order:

- To reduce the number or the complexity of the elementary steps of the process

- To limit the volume of effluents and the complexity of their treatment

- To develop a solvent regeneration process.

At this step, the availability of a simulation code will be very useful to decrease the number of experimental studies involved in these optimization studies. Pilot test, at laboratory scale, will be carried on with surrogates and genuine solutions to validate the elementary steps of the process.

Technical feasibility—The main tasks of this step will be:

- To validate all the steps of the process including ancillary operations and, notably, the solvent recycling for solvent extraction process, through pilot test in integration with other technological blocks (notably, dissolution block) and with genuine material

- To transpose the performances observed at laboratory or pilot scale at the industrial scale, taking into account not only the size effect but also the robustness of the process through time

- To choose the main device type and to do a preliminary scale up

- To do a preliminary assessment of an industrial facility.

At the end of this step, we will have a process validated at a small scale and a first laid out of the main process devices.

3.3.2.4 *Facilities and Equipment.* Several types of facilities will be necessary to perform the above research:

Basic research laboratories to carry studies on actinides and fission products' behavior in the media considered and their interaction with the different ligands examined (O-bearing, N-bearing, S-bearing. . .) using various techniques on inactive but also, as far as possible, on active material (notably important to study actinides behavior).

"Computer" laboratories for quantum chemistry approach and also for process simulation code development.

Cold laboratories equipped for basic test of various chemical media and, notably, for solvent synthesis and stability studies (irradiator device, gas and liquid chromatography coupled or not with mass spectrometer, capillary electro-chromatography. . .). A process loop will be useful for testing the miniature extractor with the different solvents considered.

Hot laboratories (gloves boxes and hot cells) equipped for process studies with surrogate and genuine solutions (distribution coefficients and transfer kinetics determination, miniature extractors. . .). A large hot cell will certainly be necessary to demonstrate the technical feasibility of the chosen process with some of its ancillary operation (solvent treatment at least).

3.3.3 Pyro Recycle Technology Enhancements

3.3.3.1 *Goals and Stakes to be Resolved During the Gen-IV Viability R&D Phase.* The major advantages of the pyrochemical method to reprocess advanced fuels, in comparison to aqueous techniques (such as higher compactness of equipment, a reduced number of transports and storage costs, but also higher proliferation resistance) are fully supporting major Gen-IV goals. It is, however, clear that the technology is far less advanced.

One of the greatest challenges will be to convert the technology into commercially viable industrial-scale processing. The goal is to understand and resolve the uncertainties in existing processes with a view to industrialization and commercial application including a complete scheme of fuel treatment, from the head end through a separation process to the immobilization of salt wastes. This includes in-depth basic chemistry, materials science, and engineering design studies.

What is known currently, what is still not known?

Even though studies on pyrochemical methods date back to the early 1960s, none of these methods have been employed on a pilot-plant scale in any country other than Russia and the U.S. Though several methods of pyrochemical reprocessing are known, only two methods based on electrochemical separations have been investigated extensively and developed to a plant scale. One is the Molten Salt Electrorefining (MSER) method for metallic fuels developed by the Argonne National Laboratory (ANL) for the Integral Fast Reactor concept, and the other is the pyroelectrochemical method for oxide fuels developed by the Russian Institute for Atomic Reactors (RIAR).

Japan's research and development activities on pyro-electrochemistry are mainly carried out by:

- Central Research Institute of Electric Power Industry (CRIEPI) with the development of a metallic fuel process

- Tokyo Electric Power Company (TEPCO), who has recently embarked in a large collaboration with RIAR to study reprocessing of MOX fuel

- Japan Atomic Energy Research Institute (JAERI), who is investigating the potential of nitride fuels for minor actinide transmutation

- Japan Nuclear Fuel Cycle Development Institute (JNC) has started an evaluation of different pyrochemical reprocessing technologies in view of future nuclear fuel cycles in Japan.

The European Commission is presently funding a project called PYROREP for partitioning and transmutation of mainly Am. Electrorefining and liquid metal extraction in molten chloride or fluoride media are the main research issues.

For pyrochemical processing to become a viable technology, certain key problems must be tackled and solved. The technology involved in pyrochemical processing is quite complex, in that the materials, equipment, and operations must be designed to stand the high temperatures. Since these electrochemical processes are to be carried out at temperatures as high as 600°C and the consolidation step for advanced fuels at $1,300^{\circ}$C, the high temperatures should be attained in remotely operable facilities.

Further development of pyrochemical processes—such as fuel reduction, electrorefining, liquid-liquid extraction, chloride distillation, metal distillation, and oxidation—are needed on the laboratory and on the engineering scale, aiming at an evaluation of the potential of this technology also from an economic point of view. Comparative assessments between dry processes and aqueous processes have to be done.

Many physico-chemical properties are not known for substances or materials that could be involved in the studied processes.

3.3.3.2   R&D Actions. The main objectives of these operations are, of course, very similar to those of aqueous processes:

- Selective extraction of the actinides for subsequent recycling; the extracted product must, therefore, meet specifications for purity with respect to the remaining non-extracted products.

- Minimization of losses of the relevant elements in the various process steps.

- Generation of process waste (not extracted elements in salt or metallic form and technological waste) compatible with a suitable treatment process.

Performed in close international collaborations, the research work should focus on the following major topics:

- Establishing a thermodynamic database on actinide-containing fuels for molten salt systems

- Testing and evaluating advanced dry reprocessing techniques

- Installing new facilities for pyrochemical reprocessing

- Developing supporting analytical techniques

- Studying innovative fuel fabrication processes in connection with pyroproccesing.

The organization of R&D should be such that at each stage a selection should take place as early as possible on the base of a balance between the foreseen advantages and the foreseen industrialization problems.

3.3.3.3   Fundamental Data. A large number of fundamental data are still missing to develop pyroprocessing technology:

- Basic thermodynamic and electrochemical data of mainly actinide elements involved in the processes to be investigated

- Vapor pressure of actinides and fission product in chloride salts, especially for the elements with high potential volatility such as Cd, Am, I

- Solubility of actinide ionic compounds (oxides, chlorides, and fluorides) in molten salts (for MSRs) or separation of elements in nuclear wastes

- Evaluation of the activity coefficients of actinide elements in molten salts

- Modeling of thermodynamic mixing of molten salts or alloys to calculate the phase diagrams and the distribution coefficient of the elements between the various phases

- Mass transfer in molten salt/liquid metal reactions or in the vicinity of an electrode.

3.3.3.4   Exploratory Studies. For the reprocessing of nuclear fuel, the following basic steps have to be considered:

- Separation of the fuel from the cladding material by mechanical techniques.

- Dissolution together with the fuel in molten salt at temperatures of the order of 500–800°C. Fluorides and chlorides are mainly proposed for this purpose although other media (phosphates, borates, nitrates, etc.) could also be considered.

- Separation of the actinides by electrorefining, transfer to a non-miscible liquid metal phase, or selective precipitation. Partitioning by extraction is based on the differences in the thermodynamic stability of compounds or complexes of these elements and on changes in their oxidation state. Extraction by a liquid metal involves stripping by means of a reducing agent. A multistage process improves, of course, the degree of refining and/or the recovery rate. In the case of electrowinning or selective precipitation, the elements are recovered by a physical method.

- Direct electroreduction or Li reduction if a metallic pyroprocess has to be applied on non-metallic fuels. The efficiency for the reduction should be tested as a function of temperature, salt bath composition, potentials, etc.

Important aspects of the processes that should be investigated are:

- Effects of impurities in the salt (e.g., fission products)

- Influence of the impurities in the atmosphere (mainly O_2 or H_2O).

3.3.3.5 ***Process Development.*** Process development provides a base for economic evaluation of a technology and its operating conditions. At this stage, the industrialization needs can be evaluated on a sound basis.

The following development step is industrialization, including:

- The development of scaled-up installations (e.g., electrorefiners capable of actinide recovery at realistic material throughputs)

- The development of computer codes to simulate the processes in different experimental set-ups for scaling-up purposes.

Although almost all industrial pyrochemical processes are batch processes, there is always a hope that continuous processes should be more efficient. Thus, it is worthwhile to have a permanent watch at possible continuous processes or partly continuous processes, when this improves the process efficiency.

3.3.3.6 ***Fuel Re-fabrication.*** Another research issue is the fuel re-fabrication. For the RIAR process, a special vibrocompaction technique is already used at semi-industrial scale. For the metallic fuel at Argonne National Laboratory West, an injection casting of the alloy has been demonstrated at pilot scale. In the case of americium-containing alloys, this method poses some problems since Am metal is relatively volatile and may evaporate. Powder metallurgy is, therefore, also being considered for the fabrication of metallic alloys. Because the cathode product contains small amounts of fission products, the fabrication requires remote handling.

3.3.3.7 ***Materials.*** A challenging research issue is also the selection of appropriate materials. Molten salts at high temperatures (500–900°C) and chlorination or fluorination processes require highly resistant materials. The goal is to increase the lifetime of existing materials used today (AlN, ZrO2, Al_2O_3, and BeO) and to find new long-lasting materials, where possible.

The oxide processes use pyrolytic graphite as cathode and crucible material. This material, which is resistant to air oxidation up to 500°C, is expensive and indigenous sources must be sought or alternative materials must be found in view of process industrialization.

3.3.3.8 ***Head- and Back-End.*** It is evident that pyrochemical processes include not only a "core" process, in which the recyclable elements are extracted, but also numerous ancillary operations at the front end or the tail end of the main process, where research and development are also mandatory:

- Prior preparation of the feed stream to ensure compatibility with the core process (e.g., de-cladding of solid fuel, halogenation of oxide fuel, and dissolution in molten salts)

- Purification of the output stream with recycling of contaminants back into the core process and treatment of the process and conditioning of waste materials in accordance with their chemical and radiological properties (e.g., vitrification)

- Decontamination of technological waste (e.g., crucibles) with recycling of contaminants back into the core process and conditioning of the actual waste materials (e.g., melting of scrap metal)

- Decontamination of all process off-gases with recycling of contaminants back into the core process.

The performance characteristics required for the core process and for the ancillary operations depend directly on the losses of recyclable elements into the waste stream and on the desired degree of purity of the recycled elements.

If a voloxidation of the spent oxide fuel is included in the pre-treatment, several fission products are distilled off including ^{129}I, which could be collected with 99.9 % yield. During the electro-refining process, 95% of the ^{99}Tc is collected together with the noble metal fission products in the bottom Cd layer of the electrolysis cell. For the transmutation of the two fission products, special targets have to be fabricated.

The waste produced throughout the process is expected to be minimal, since most of the materials (such as the eutectic salt mixture, the cadmium and bismuth metal, and the chlorine gas used for the chlorination) will be recycled. The LiCl-KCl mixture can be recycled after purification, while the salts of fission products are electrolytically decomposed and converted to oxides for vitrification in borosilicate glass. Non-recyclable waste will be solidified directly into a synthetic rock such as zeolite or sodalite with high mechanical integrity and leach resistance.

Significant work on electrorefiner salt cleanup and high-level waste form production needs to be done in order to achieve the very high actinide recoveries (~99.9%) that are the objective of the process—developing any secondary waste stream treatment that may become necessary to achieve this recovery goal. Also, it is necessary to complete certification of the two high-level waste forms (metal and ceramic) for repository disposal.

Safeguards implications of an experimental pyroprocess facility operated in a fuel recycle mode (building on the current waste management or spent fuel treatment mode) must be investigated.

3.3.3.9   *Installations.* At ANL laboratories in the U.S., existing facilities should be used to further develop the metallic pyroprocessing technology. In RIAR (Dimitrovgrad in Russia), development of the oxide pyroprocesses will be continued and at JNC in Japan, a new laboratory is being set up to compare the performances of various technologies Also, in India at the Indira Ghandi Research Center, a pyrochemical laboratory has been in operation since the late 1990s. Dedicated facilities exist in the European Union, though the number of hot pyrochemical facilities is limited. At AEAT, BNFL, ITU, and CEA, such facilities are under construction or are already in operation at present.

In the long term, these installations should provide the fundamental data on process chemistry and engineering to provide a firm basis for selecting the most appropriate concepts and to construct a pyroprocessing facility at industrial scale.

What are the costs involved?

Cost figures can be only very indicative because, as shown above, many reprocessing options are under investigation and the R&D needs involved can differ significantly between the various routes. The pyroprocessing technology will, in any case, require a consistent amount of basic research and process development. Technology development could run for 10 to 15 years. Indicative costs in millions of dollars are given below.

Basic research	200–400M
Process development	300–600M
Prototype demonstration	200–300M

3.3.4   Recycle of Particle Fuels

3.3.4.1   Composite Fuels Treatment/Goals and Stakes. Among the many criteria put forward to define the fuel for future nuclear systems, several ones (fission products retention, conductibility, actinides contents, thermodynamic stability) lead the engineer to propose fuels that combine a high-density-actinides compound (carbide, nitride, oxide) coated with or embedded in a highly refractory ceramic material (Figure 8).

Whatever is the structure, one ends up with a local structure where the tiny actinides compound grains are totally surrounded by the refractory material. Thus, when speaking of fuel treatment to recover and recycle the actinides, problems arise regarding the accessibility of the elements of interest as well as the inert material back-end management.

A mechanical treatment to break the inert material to free the actinide compound grains may prove to be not efficient enough for a quantitative actinides recovery due to the average grain size, which ranges from tens of micrometers to 1 millimeter. It is, however, a prerequisite before going into dissolution and purification schemes. The biphasic micro or milli-structure leads to consideration of two types of treatment:

- *Homogeneous treatment*—The two materials, the actinides compound and the inert, are dispersed in a given solvating media. Then a suitable method is used to isolate and recover the actinides.

- *Heterogeneous treatment*—The inert material is first selectively eliminated by a chemical or a physical mean, thereby leaving the way to access the actinides for further treatment (dissolution and separation of the fission products).

In both cases, it is of primary importance to deal with a refractory material that was precisely chosen for its physical and chemical stability.

- Regarding the inert material management, three routes must be evaluated:

- Conditioning with the fission products

- Separation and specific conditioning

- Recycling to the re-fabrication process.
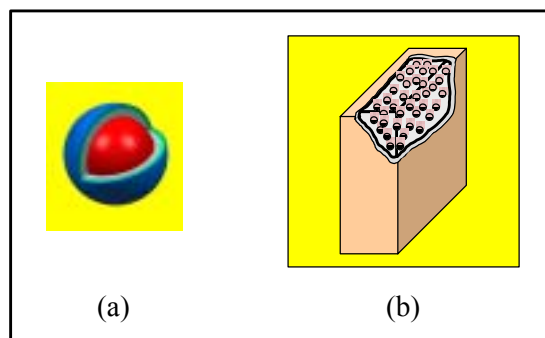
Figure 8. Example of fuel composite structure (a) coated particle, (b) actinides dispersion in inert material.

3.3.4.2   State of the Art. The most promising refractory materials that are proposed, depending upon the nature of the actinides compound, seem today to be the following:

- SiC, ZrC, or TiC for the actinides carbides

- TiN for the actinides nitrides

- MgO for the actinides oxides.

The silicon carbide (carborundum) is often associated with graphite such as with the TRISO particles fuel of the HTR reactor.

Regarding the physical techniques to treat the material, only the burning of the graphite and the crushing of the carborundum have been studied in the past, most notably in connection with PBMR development. Results were not totally satisfactory; recovery yield of the actinides hardly exceeded 95 % and it involves quite complex technologies particularly for the off-gas treatment. No other physical method has been studied to such extent.

The literature offers some chemical alternatives to attack such materials, ranging from mineral acids and bases to gaseous reactants such as hydrogen fluoride. However, data on performances (yield and kinetics) are scarce.

3.3.4.3   Possible R&D Routes. Taking into account the very innovative features of the subject, a four-step development program can be proposed:

*Selection of concepts—*Having a broad and open approach, several candidate methods for accessing the actinides compound should be investigated. Chemical methods should be examined: carbochlorination, plasma fluorination, hydrofluorination, chlorination, and direct electrochemical dissolution in a molten salt. Physical methods are also of interest: crush/burn techniques and ultrasound de-structuration. Experiments, mainly at laboratory scale on inactive material, would focus on efficiency and simplicity. The work would also involve basic research on the very fundamental chemical properties of these refractory materials in order to seize their interactions with chemical reagents.

Still speaking of the inert material, assessment of possible management routes would need to (1) select, evaluate, and possibly test processes for its separation (and possibly its recycling) and (2) then define a confinement matrix that allows long-term disposal. In addition, its incorporation in the fission products confinement matrix should be assessed.

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Validation of concepts—Having chosen some few promising head-end methods, it would be necessary to perform tests in more realistic conditions in order to take into account the behavior of the actinides and the fission products. Depending on the availability of representative materials, experiments would be carried on with non-irradiated fuel elements or surrogates.
Thus, the final selection of a process could be made considering criteria that are specific to fuel reprocessing, in particular:

- Intrinsic safety

- Availability of a An/FP separation process in the chosen media

- Compatibility with a waste confinement material and process.

Process development—As explained above, the chosen process may not be a classical room temperature dissolution with a "soft" reactant like nitric acid. Dissolvers like the ones used in today plant would certainly be inadequate and specific equipment should be developed to take into account:

- Corrosion matters

- Fuel element geometry

- High temperatures or pressures, if necessary

- Volatile compounds management

- Residues (if any exist)

- Reactant recycling.

Regarding the inert material back-end management, separation process development should be included in the overall separation process development. If a specific conditioning is chosen, a manufacturing process has to be developed and tested. The long-term behavior of the produced confinement matrix should be characterized.

Technical feasibility—Before implementation into a pilot- or industrial-scale plant, it would be necessary to validate the process, including ancillary operations, with genuine material and using devices prototypic of industrial applications. This step could be run in integration with other primary technological blocks (separation, re-fabrication) but not necessarily.

3.3.4.4 *Facilities and Equipment.* The work will involve several types of facilities:

- Cold laboratories equipped for basic test of various chemical scheme (high-temperature conditions, corrosive products handling) and physical dispersion methods: grinding, plasma torch, and ultrasonic devices.

- Basic research laboratories to carry studies on solid-liquid and solid-gas interactions using various spectroscopy and electrochemistry techniques.

- Hot laboratories (gloves boxes and hot cells) equipped with some of the above apparatus, mainly high-temperature devices and the possibility to handle corrosive gases. A large hot cell will

certainly be necessary to implement not only the heart of the chosen process, but also all the ancillary operations to be able to demonstrate the technical feasibility.

- Equipment development facility allowing the size up and test of innovative apparatus.

3.4   Concept-Independent Front End Fuel R&D

Development of technology for the concept-independent front-end links of the fuel cycle (exploration, mining, enrichment) is essentially stagnant in light of the current glut of uranium and enriched uranium in storage. For example, exploration in Canada is carried out only to the degree necessary to comply with lease requirements. Enrichment technology development has stopped in the U.S., is phasing out in France, and is continuing in Australia.

Nonetheless, the scenario results from Chapter 3 of the FCCG main report showing that forward fueling requirements for once-through Gen-IV systems deployed after ~2030 will exceed the Redbook Known plus Speculative reserves of ~15 million tonnes U recoverable at <130 $/kg.

While in light of the more pressing back-end issues, the FCCG does not put strong stress on front-end R&D during the initial Gen-IV viability phase; such R&D should, in fact, receive attention starting in the Performance R&D phase. Three areas of specific focus are discussed here:

- Mining method improvements

- Ongoing evaluations of potential for ocean extraction

- Enrichment technology.

3.4.1   Improved In Situ Leaching Technology

Uranium supply is a crucial matter for sustainability of nuclear energy for the near future. According to the IAEA projection, the supply to meet the requirement of nuclear fuel is not optimistic after 2030. In order to avoid a shortage of uranium resources foreseen in the projection, more efforts to discover large-scale uranium deposits are required. From the point of view of technology development, mining technology should be improved toward more economic and environment-protective manners. The in situ leach (ISL) mining method is a key technology to solve the issue of uranium shortage in the future.

3.4.1.1   Background. In situ leach extraction of uranium accounted for about 15% of worldwide uranium production in 2000, and new ISL projects under development in Australia, Kazakhstan, and Russia will ensure an important role for ISL in the future. Though quantitatively much less important, other leaching technologies—including heap leaching and block or stope leaching—are being used to supplement conventional mining and milling and to improve the economics of uranium recovery from low-grade ore. In situ leach is currently limited to recovering uranium from sandstone deposits, while other forms of leaching can be used on ores associated with a wide variety of geologic rock types.

3.4.1.2   Economics of ISL. In situ leach is typically a lower-cost alternative to underground or open pit mining and conventional milling, which allows economic exploitation of lower-grade resources. In addition, other leaching methods can be used to process low-grade ore that would be otherwise economically unattractive. In situ leach also has environmental advantages over conventional mining and milling of uranium in that it minimizes surface disturbance. However, ISL is not without potential environmental consequences, particularly its potential impact on groundwater resources. There are currently no reliable estimates of the percentage of worldwide uranium resources that are amenable to ISL

extraction. However, ISL-amenable resources account for nearly all of the reasonably assured resources (RAR) recoverable at a cost of <USD40/kgU in Kazakhstan and Uzbekistan, which together account for about one-quarter of worldwide reasonably assured resources in this cost category. Similarly, Australia, Russia, and the U.S. also have significant ISL-amenable reasonably assured resources recoverable at <USD40/kgU.

The economic advantages of ISL are in part attributable to the relatively shallow depths of current operations. In addition, the poor water quality of the leach aquifers in Australia, Kazakhstan, and Uzbekistan currently preclude the need to restore the aquifers to pre-mining conditions, which gives these operations an economic advantage compared to those in the U.S. where aquifer restoration is required.

3.4.1.3 *Deep Mining.* As we look to the future, new ISL operations will increasingly involve deeper deposits. This is already evident by new ISL projects that are being developed in Kazakhstan and Uzbekistan that involve deposits at depths below the surface of between 600 and 700 m. Similarly, pilot leach tests have been conducted on deposits in Russia and the U.S. at comparable depths. The ISL operations at increasing depths represent both technical and economic challenges. Increased depth equates to increased costs of well construction. The depths being contemplated will test the strength of equipment currently in use and may require more expensive, higher-strength pipe, greater capacity pumping equipment, etc. The greater depths may require higher leaching cutoff grades and broader spacing between injection and production wells to remain economically attractive. At the same time, greater depth is accompanied by increased temperatures in the host aquifer, which may increase leach efficiency and shorten the time needed to deplete the resource and partially offset some of the negative aspects of deeper leaching.

The ISL-amenable deposits comprise an important part of the worldwide uranium resource base. Since future development of ISL resources will involve deeper deposits, research is required into the technical and economic impact of greater operating depths on ISL operations. The initial basis for this research will be compilation of current operating experience at different depths:

- What equipment works at what depth?

- What wellfield spacing is required to offset higher well construction costs?

- How will increased spacing requirements affect leaching efficiency?

- How does temperature affect leach reaction time and solubility of leach solutions, and what are the economic consequences/benefits of this relationship?

In addition to compiling actual operating experience in these areas, basic research should be completed to at least establish theoretical limits on strength of materials at depths greater than they have been previously used. Similarly, theoretical estimates should be established for other ISL operating parameters at greater depths.

The trend for ISL operations worldwide will be toward developing deeper deposits. Therefore, research should concentrate on deeper ISL operations and economics. At the same time, there are shallow sandstone deposits close to or even above the water table, which, except for insufficient hydrologic head, would seem to be amenable to ISL extraction. Limited research, including pilot leach testing, has been done in which an artificially higher water table has been created by water injection, followed by ISL extraction of uranium. Further research is warranted on the potential for ISL operations in unconfined aquifers with limited hydrologic head. If successful, this research could open up significant shallow,

low-grade uranium resources that are not economically attractive for open pit mining and conventional milling. Included in this research should be:

- Solubility of leach solution components (CO_2, O_2, H_2O_2, etc.) with increasing hydrologic head

- Aquifer restoration at shallow depths.

3.4.1.4   *Ecological Effects*. Aquifer restoration to baseline conditions is currently only required in the United States. The ISL operations elsewhere in the world are being conducted in aquifers where the water quality is unsuitable for human or agricultural use because of high salinity and radionuclide contamination. The technology for aquifer restoration is well established by pilot tests in the United States. There is, however, limited information available on aquifer restoration under large-scale commercial conditions—how effective is it, how much does it add to total production costs, and how long does it take. Studies of self-remediation of aquifers where acid leach systems have been used in Kazakhstan suggest that this process can take between 30 and 40 years.

While the majority of ISL operations do not now require aquifer reclamation, we cannot assume that this will always be the case. The ISL-amenable resources occur outside the United States in aquifers that are suitable for either human or agricultural use. Therefore, more research on ISL aquifer restoration needs to be completed. Operating histories need to be compiled; successes and failures need to be documented. As operations move to greater depths, the potential for failures in well construction increase, which in turn increases the potential for contamination of overlying and underlying aquifers. The Czech Republic projects that it could spend in excess of USD1 billion between 1999 and 2040 to decontaminate aquifers contaminated by ISL operations. There is much to learn from this example, both as to avoidance and mitigation of the problem.

3.4.1.5   *Low-Grade Ore Mining*. In situ leach is by far the most important of the uranium leaching technologies. However, surface heap leaching and underground stope or block leaching also have potential to more economically and efficiently exploit the uranium resource base. These leach technologies have the potential to process ore that is too low grade to be economically processed in a conventional mill. Without the potential of these technologies, this low-grade ore may either be left behind during conventional mining operations or stockpiled as uneconomic waste.

Heap leaching and block/stope leaching are currently being used in China and Russia and have been extensively used elsewhere in the world in the past. Therefore, case histories are the first place to start in evaluating the potential for these extraction technologies to enhance the efficient use of the uranium resource base. Questions to be answered in this research include:

- Heap Leaching

    - Heap construction—dimensions, ore particle size, pregnant acid collection system
    - Acid delivery system—effectiveness of top acid delivery compared to intra-heap delivery systems
    - Leaching strategy—changing acid strength over time
    - Processing of pregnant liquor—ion exchange compared to solvent extraction
    - Efficiency—overall recovery factor; recovery versus time curve
    - Economics.

- Underground Stope or Block Leaching

    - Stope preparation—size, blasting techniques; optimum ore fragmentation

    - Acid delivery and collection systems

    - Potential environmental consequences and mitigation procedures

    - Processing of pregnant liquor—ion exchange compared to solvent extraction

    - Efficiency—overall recovery factor; recovery versus time curve

    - Economics.

The ISL extraction of uranium and to a lesser extent surface heap leaching and underground block/stope leaching are important to the efficient exploitation of the uranium resource base. In particular, these leaching technologies provide economic access to lower-grade uranium resources. Many of the shallow deposits amenable to ISL extraction are rapidly being depleted, and the industry is faced with development of deeper deposits. Increasing costs and technical uncertainties are the result of increasing depth.

Research into the technical, economic, and environmental aspects of deeper exploitation by ISL extraction will help ensure that it remains an important extraction method, and in turn that lower-grade resources will remain economically attractive. Similarly, research into surface heap leaching and block/stope leaching could enhance their expanded utilization for recovery of uranium from otherwise uneconomic low-grade ore.

3.4.2   Uranium from Seawater

The oceans contain various kinds of substances in solution and particularly uranium at a concentration of 0.003 mg/L. Therefore, seawater is a potential reserve of 4 billion tons of uranium, which is theoretically an inexhaustible resource for uranium.

Uranium is assumed to be present in seawater as uranyl tricarbonate ion, $UO_2(CO_3)_3^{4-}$. The minimal energy necessary for the extraction of one mole of uranium from seawater has been evaluated to 46 kJ/mole (which is much less than the fission energy of one mole of uranium: see Note 1 at the end of the document for further details).

3.4.2.1   Basic Methods of Uranium Extraction. The Atomic Research Establishment in Harwell (AERE), United Kingdom, carried out the first studies on uranium extraction from seawater as early as 1953. Extensive efforts have been made in Japan since the early 1960s. In the Federal Republic of Germany, continuous research on the subject has been in progress since about 1973. Investigations have also been carried out in France, Italy, Russia (former Soviet Union), Finland, India, and more recently in the U.S. and Sweden.

A pumped system was first proposed with the necessity to handle enormous volumes of seawater to extract uranium from seawater. Assuming the adsorption efficiency of 30%, $1 \cdot 10^{12}$ m^3 of seawater a year ($2.7 \cdot 109$ m^3 a day) must be contacted with the adsorber to recover about 1,000 tons uranium per year. One can evaluate that energy of $9,81 \cdot 10^4$J (or 0.027 kWh) is spent to pump 1 m^3 of seawater at 10 m high. From theses figures, it is easy to calculate that the pumping energy would be about 60% of the electrical energy recovered from uranium used in LWR (for more details, see Note 2). Thus, if we examine the feasibility of such a scheme (involving the pumping of seawater) in terms of energy, it

becomes clear that the energy expended for pumping is too large compared to the energy recoverable from uranium from seawater: the recovery of uranium from seawater becomes meaningless.

Another concept bound with desalination of seawater has been studied. The idea consists of coupling recovery of uranium from seawater and desalination process, which mobilizes large amounts of seawater. Unfortunately, a problem of size is raised one more time. Indeed, in Spain, for example, to produce 50,000 to 100,000 m^3 of fresh water a day, an installation treats 75,000 to 150,000 m^3; that is to say that one unit moves about 250 to 500 gU/day, or 90 to 180 kg U/year, which is 1,000 times less than the needed quantity to feed only one LWR.

A system using the turbo pumps supplying the cooling circuit of a nuclear plant has been imagined. For a reactor of 1,000 MWe, with a water post supplying seawater by 2 turbo pumps of 2,725 t/h, the annual flow of seawater is about 40 million m^3, equivalent to 130 kg U a year. This shows the inapplicability of such a process.

At last, because of the impossibility of pumping, some have proposed to use natural sites where the magnitude of tides could fill tanks of seawater. A hundred of such sites have been identified in the world, among the UKEA project, near Anglesey Islands. However, this concept has not been pursued.

Thus, there is a consensus that static systems using natural currents with more particular adsorption methods are the most applicable for the recovery of uranium.

One of the most important aspects in establishing the recovery process is the development of an adsorbent with high uranium selectivity and a high rate of uranium adsorption. The results of many investigations with inorganic adsorbents have indicated that hydrous titanium oxide and composite adsorbents (with hydrous titanium oxide and other materials) are the best potential candidates. Table 2 represents the uranium loading of selected inorganic sorbents.

However, the rate of adsorption of the uranium on granulated hydrous titanium oxide or on titanium-activated carbon composite adsorbent was found to be only about 0.2 mg/g adsorbent in 10 days; moreover, the mechanical stability was insufficient. This shows the need to develop a new adsorbent with a higher adsorption rate.

Therefore, extensive investigations have been made on organic sorbents, and the main results are shown in Table 3.

More attention has been given to the organic polyacrylamidoxime (PAO) because of its higher sorption capacity and its outstanding mechanical stability.

Table 2. Uranium loading of selected inorganic sorbents.

Sorbents	Uranium Loading	References
Hydrous aluminum oxide	61 μg/g Al	1965 AERE
Hydrous iron (III) oxide	60 μg/g Fe	1965 AERE
Silica gel	27 μg/g sorbent	1977 IUPAC Congress
Hydrous lanthanum oxide	38 μg/g La	1965 AERE
Hydrous titanium oxide	200 μg/g Ti, 200 μg/g sorbent	1968 Nuclear Energy Society
Hydrous titanium oxide (freshly precipitated)	1,550 μg/g Ti	1971 Nippon Kaisui Gakkai-Shi
Basic zinc carbonate	540 μg/g Zn	1965 AERE
Hydroxide tinoxide	17 μg/g Sn	1965 AERE
Hydrous zirconium oxide	13 μg/g Zr	1965 AERE

Table 3. Uranium loading of selected organic sorbents.

Sorbents	Uranium Loading	References
Polystyrene acetylene phosphoric acid	24 μg/g sorbent	1965 AERE
Resorcinol arsonic acid	1,112 μg/g sorbent	1965 AERE
Formaldehyde copolymer	45 μg/g sorbent	1987 Sep. Sci. Technol
Polyacrylamidoxime	3,600 μg/g sorbent	1983 Chemiker Zeitung 1982 Zeitung Naturforsch
Oxamidoxime terephtalic acid	240 μg/g sorbent	1975 Japan Kokai
Polyglycoaltriaminophenol	45 μg/g sorbent	1977 IUPAC Congress
Hyphan on cellulose	80 μg/g sorbent	1977, Fresenius Z. Anal. Chem.
Macrocyclic hexacarboxylic acid	70 μg/g sorbent	1980, J. Am. Chem. Soc.
Macrocycloimideresin	930 μg/g sorbent	1981, Bull. Soc. Seawater Science
PTO	50 μg/g sorbent	1981 Synthesefasern, Verlag Chemie, Weinheim

Some properties of the PAO granules are summarized below:

Particle size (for wet granules)	0.25–0.4 mm
Water retention capacity	73.2 wt- %
Bulk density	0.87 g/ml
Specific gravity	1.1 g/cm^3
Pore volume	0.82 cm^3/g
Mean pore radius	250 A°
Specific surface	150 m^2/g
Loading (g/metal/kg resin)	3.60
Concentration factor ($\frac{g\ metal\ /\ kg\ resin}{g\ metal\ /\ l\ sea\ water}$)	1.1×10^6

Description of the main present concepts of systems for the recovery of uranium from seawater. Principle of extraction:

The main steps of the synthesis of the adsorbent material are shown below.

Synthesis of Sorbent Material

Non woven high polymer material

↓

Electrical irradiation

↓

Graft polymerization

↓

Amidoxime resin

↓

Extraction of Useful Materials

Immersion

↓

Elution and Refinement

Elimination of seaweeds and shells

↓

Elution of alkaline metals

↓

Elution of uranium

↓

Refinement by separation

Regeneration of sorbent materials

↓

- Adsorbent material: PAO resin

Non-woven high polymer → Polymerized material → Sorbent material PAO

Graft polymerization by irradiation

Functional radical hydroxylamine

A non-woven material, mainly made of polyethylene, is subjected to electrical irradiation (2Mev). With the introduction of acrylonitril, a graft polymerization by irradiation occurs. Then, the cyano radical from acrylonitril reacts with hydroxylamine and is transformed into amidoxime.

Two different methods are proposed for the graft polymerization: (1) liquid phase grafting or (2) monomer impregnated grafting. In the first one, the starting irradiated material is plunged into the monomer solution. This process is useful for increasing the capture ratio of useful metals. The second method, which is under consideration, is adapted for high-retention power materials and consists of the impregnation of only the necessary amount of monomer in the irradiated material.

- Conceptual design plants:

    Hypothesis:

Amount of recovered uranium	1,200 t/year
Adsorbent material performance	6g U/kg material
Length of one extraction	60 days
Number of extractions by year	5
Number of consecutive uses of the adsorbent material	20

The plant is composed of one unit on earth used for the separation by elution and one floating unit on sea. The adsorbent is concentrated in special recipients and immerged during a determined time. Then, it is recovered and transferred on the earth unit, where it is cleared of silt and algae.

- *Extraction system*—The adsorbent material is rolled up and placed in 30 × 15-cm cassettes. Each cassette contains 120–140 layers of adsorbent. These cassettes are put in circular metallic cages, also called adsorption beds. Each adsorption bed is made up of 144 cassettes; its thickness is about 15 cm and its diameter 4 cm, and it contains 125 kg of material. The adsorption beds are fixed together with ropes and separated from each other by a space of about 0.5 m; this constitutes a "base unit".

- *Fixation system*—Three different mooring methods have been studied: (1) buoy system, (2) floating body system, and (3) chain binding system.

    1. The buoy system: Each unit, made of 100 adsorption beds, is fixed to a buoy. The buoys are attached together with ropes to anchors. After 60 days, the units are raised with crane ships, with a capacity of 1,000 tons each.

2. The floating body system: Each barge supports 540 units (each unit is composed of 100 adsorption beds). Cranes placed on barges are used to raise units and are then charged on recovery ships. These ships can be used 300 days a year.

3. The chain binding system: In this process, the units, composed of 10 adsorption beds, are fixed to a chain binding system at a depth of 50 m. The chain binding system is divided into 14 independent systems, with a length of 23 km each. Each segment, constituted of 2,300 units, is fixed to a drive system, which is maintained at a constant depth with the help of buoys (placed every 150 m). To recover the units, the chain binding system moves as a ski lift and is transported back up on the earth base to unhook the units. With a cycle of 60 days, 134 units a day can be treated. The main characteristics of each system are given in Table 4.

- *Elution*—Once raised, the adsorption beds are cleared from algae and silt. Then, they are plunged in 0.01 molar hydrochloric acid to separate alkali and alkaline earth metals. To isolate heavy metals, the beds are placed in 0.01 molar nitric acid. Liquids containing useful metals flow through a canalization, which is charged in chelating resin, able to fix these metals. The semi-finished product is then refined to extract uranium on oxide form. At last, an alkali treatment with KOH solution is used for the reconditioning of the adsorbent.

Another method, which consists in a fractionated elution for the different metals, is possible; the difference is that 0.5 molar hydrochloric acid is used for the elution of useful metals. The fractionated elution, still under consideration, would be able to reduce the elution time and to increase the elution efficiency (96.5% instead of 80%).

3.4.2.2   Preliminary Cost Studies. Some cost evaluations have been carried out (Table 5). The main results are summarized below (figures given for a yearly production of 1,200 tons of uranium).

Table 4. Main characteristics of the buoy system, floating body system, and chain binding system.

Buoy system	Adsorption beds	5,653
	Cylindrical buoys	4,530
	Recovery ships	59
	Necessary surface	180 km^2 (4,530 buoys separated with 200-m spaces)
Floating body system	Adsorption beds	4,800
	Barges	6
	Recovery ships	20
	Necessary surface	13 km^2 around each barge, total: 80 km^2
Chain binding system	Adsorption beds	38,400
	Buoys	2,142
	Mooring supports	42
	Drive systems	14
	Cables	14
	Necessary surface	150 km^2

Table 5. Preliminary cost studies.

Cost of the fabrication of the adsorbent material (liquid phase grafting)	39.67 million dollars/year	
Cost of elution and refining	68.68 million dollars/year	
Cost of installation on seawater of the adsorbent material	Buoy system	457.12 million dollars/year
	Floating body system	422.51 million dollars/year
	Chain binding system	213.27 million dollars/year
Extraction cost	Buoy system	450.68 dollars/kg U
	Floating body system	421.71 dollars/kg U
	Chain binding system	241.43 dollars/kg U
Repartition of the extraction cost	Buoy system	Fabrication of the material: 10% Installation in seawater: 85% Elution and refining: 5%
	Floating body system	Fabrication of the material: 10% Installation in seawater: 84% Elution and refining: 6%
	Chain binding system	Fabrication of the material: 17% Installation in seawater: 73% Elution and refining: 10%

Remarks: The building redemption is 30 years. The installation redemption is 15 years.

Thus, the cost of uranium extracted from seawater with topical technique is about 5 to 10 times more expensive than the price of uranium extracted from mines (from 52 to 65 $/kg U).

3.4.2.3   Research and Development Needs. Extensive R&D works are required for the following.

- Adsorbent material:

  - Increasing the surface of contact between the adsorbant and seawater and studying immersion methods that could improve this contact

  - Producing a material with a stable quality

  - Developing the process of polymerization by impregnation

  - Studying the process of preconcentration by nanofiltration.

    Here, the ocean current would pass through an ordinary filter on the buoy, entering a region bounded by nanofilter film. The function of the nanofilter is to allow the common metal ions to seep out into the ocean, while blocking outflow of the heavy metals. Nanofiltration could concentrate uranium with a factor 100 and also makes seawater less acid (from a pH of 8.3 to 9.5). This makes possible the use of more selective adsorbent materials, essentially "trained" during the polymerization on template ions of the type to be separated. For such material, it is expected that each kg of adsorbent would retain as much as 90 g uranium. Thus, research on new types of resins, as polyphenolcoformaldehyde, template, polyresorcinol, or DTPA resins needs to be continued.

- Improving its performance.

    It has been proven (JAERI 1998) that amidoxime resin created by cografting polymerization with acrylonitril (AN) and methacrylic acid is 5 times more adsorbent than with AN. So, further investigations are needed for optimizing the composition of the mixed monomer solution for the cografting reaction and for reducing the amidoximization time.

- -Increasing its length of life in order to raise the number of uses

- **Extraction system:**

    - Carrying out a global evaluation of the more appropriate shape for the cassettes containing adsorbent

    - Studying a method for the transport of these cassettes on earth

    - Decreasing the weight of the metal cages that contain the adsorbent material (as using plastic cages or letting the adsorbent float through the current without any cages)

    - Evaluating bending and tension fatigue of ropes used in the equipment: since the end of June 2000, nylon ropes, which have excellent characteristic against bending fatigue, have replaced polyethylene ropes.

- **Fixation system:** here are listed the main problems to solve for each system

    | Buoy system | • Dynamic characteristics of the units under bad weather (more particularly, influence on the recovery work) |
    | | • To increase the recovery speed of the adsorption beds |
    | | • To reduce the necessary surface |
    | Floating body system | • Dynamic characteristics of the units under bad weather (contacts and hurts between base units) |
    | | • To reduce the weight of barges by optimization of the structure |
    | Chain binding system | • To prevent the unhook of cables near the mooring supports under bad weather |
    | | • To reduce the amount of cables |

- **Elution and refining:**

    - Trying to find an elution process, which does not degrade the efficiency of extraction: some studies have been carried out to develop an elution process based on the use of organic acids

    - Studying and improving the contact between eluent and cassettes

    - Developing the fractionated elution method

- Studying processes in which the eluent flows out parallel to the adsorbent material (if the eluent flows out perpendicular to the sorbent, problems of flow reduce the elution efficiency).

- Environment: studying the consequences of the exploitation of such a system of extraction on the seawater wildlife and flora.

3.4.3   Conclusion

The aim of this document was to realize a synthesis on the status of art and to show the present main problems to solve concerning the extraction of uranium from seawater. By studying all the interrogations still raised, and thanks to the present technology, it could be possible, even if it may not be used before 50 years or more, to know if, as Richard L. Garwin said: "uranium is from seawater a green fuel for the future".

3.4.3.1   *Minimal Energy to Extract One Mole of Uranium from Seawater.*

Definition:

$$S = k \log N, \text{ with } S: \text{ entropy}$$
$$k: \text{ constant}$$

Variation of entropy created by the concentration process (also called mixture entropy):

$$\Delta S = k \,\eta\, \log C, \text{ with } \eta: \text{ Avogadro constant}$$
$$C: \text{ concentration factor } (C = 1/\;3 \cdot 10^{-9} = 3 \cdot 10^{8})$$

If the process is reversible:

$$\Delta Q = T\Delta S = kT \,\eta\, \log C, \text{ with } k = \text{Boltzmann constant}$$
$$T = \text{temperature (Kelvin) (25°C)}$$
$$= 1.38 \cdot 10^{-23} \cdot 298 \cdot 6 \cdot 10^{23} \cdot \ln\left(3 \cdot 10^{8}\right)$$
$$= 48 \text{ kJ/mole}$$

3.4.3.2   *Energy for Pumping.*

- $1 \cdot 10^{12}\, m^3$ a year must be treated to recover 1,000 t U a year, with an adsorption efficiency of 30%

- 1 t uranium $= 10,000$ tep $= 45,000$ MWh

- 1,000 t U = 45,000 GWh = electrical energy obtained in a LWR

- $9.81 \cdot 10^{4}\, J = 0.027$ kWh are necessary to pump 1 m^3 of seawater at 10 m high.

- For pumping $1 \cdot 10^{12}\, m^3$ of seawater at 10 m high, an energy of

- $9.81 \cdot 10^{16}\, J = 27,000$ GWh is required

- The ratio between the energy required for pumping and the electrical energy obtained from uranium in LWR is: $27,000/45,000 \approx 60\%$.

4.   LIST OF FUEL CYCLE
RELEVANT TESTING FACILITIES/AVAILABILITY

4.1   Worldwide R&D Infrastructure

4.1.1   R&D-Infrastructure

The deployment of nuclear energy will demand the development and industrial upscaling via prototype/pilot plants of new technologies currently under development.

The R&D and the facilities for its execution are a necessary factor to develop these new technologies. This section will address this issue and will list, based on information publicly available, those facilities available today and planned in the near future. A list of possible gaps or problems in availability of
R&D-infrastructure will be proposed as conclusion of this note.

The main limitations in availability of infrastructure for fuel-cycle-related R&D are to be seen in essentially four categories:

1.   Fabrication of new fuels/targets needing essentially hot-cells for the handling of these materials;

2.   Irradiation facilities to test these fuels/targets in representative conditions;

3.   Reprocessing facilities to test the basic science and technological development of wet and dry reprocessing techniques

4.   Post-irradiation facilities able to handle very-active fuels/targets.

These are for the fuels and targets foreseen in the Gen-IV nuclear energy system development:

•   (High burnup) UOX and MOX

•   HTGR fuels

•   High TRU-content fuel including IMF, MA targets, and FR fuels for use in different types of reactors.

Closed fuel cycles mostly envisage having a very close integration between fabrication, irradiation and reprocessing of the fuels, and the above categorization of R&D facilities may therefore not be considered as independent categories or unlinked facilities. A very strong interaction between the R&D teams and most probably also between the infrastructure itself is to be considered. We will address in the following sections essentially the need for hot cells (i.e., needed for fabrication and reprocessing research including PIE) and the need for irradiation facilities.

The following tables will list the main irradiation facilities (Table 6) and fuel/target fabrication, reprocessing and PIE R&D facilities (Table 7) that are currently available for R&D. Table 8 lists the facilities for basic science aspects and critical assemblies for integral experiments. These tables also indicate links to websites (if available) with extensive detailed information on these facilities. The reader is referred to these links for further information.

Table 6. Overview of today's available main irradiation facilities of potential interest for Gen-IV R&D (based on IAEA RRDB, Reference Data Series No. 3).

Country	Name	Criticality Date	Power (MWth)	Thermal Flux (n/cm².s)	Fast Flux (n/cm².s)
Australia	HIFAR	1958/01	10	1.40E+14	3.50E+13
Belgium	BR2	1961/06	100	1.00E+15	7.00E+14
Canada	NRU	1957/11	135	4.00E+14	4.50E+13
China	HWRR-II	1958/09	15	2.40E+14	5.20E+12
	HFETR	1979/12	125	6.20E+14	1.70E+15
Czech Republic	LVR-15	1957/09	10	1.50E+14	3.00E+14
Denmark	DR-3	1960/01	10	1.40E+14	4.00E+13
France	OSIRIS	1966/09	70	2.70E+14	2.60E+14
	CABRI	1963/01	25	1.90E+14	1.50E+17
	ORPHEE	1980/12	14	3.00E+14	3.00E+14
	PHEBUS	1978/08	40	2.80E+14	5.00E+13
	SCARABEE	1982/01	100		1.20E+15
	PHENIX	1973/08	56.3		7.20E+15
	JHR	*			
Hungary	Budapest RR	1959/03	10	2.50E+14	1.00E+14
Japan	KURRI	1964/06	5	6.00E+13	6.50E+13
	JMTR	1968/03	50	4.00E+14	4.00E+14
	JOYO	1977/04	100		2.70E+15
	MONJU		714		
	HTTR	1998/11	30	7.50E+13	2.00E+13
Korea	HANARO	1995/02	30	4.50E+14	3.00E+14
Netherlands	HFR	1961/11	45	2.70E+14	4.60E+14
Norway	HBWR	1959/06	20	1.00E+14	1.00E+14
Poland	Maria	1974/12	30	4.50E+14	1.00E+14
Russia	MIR M1	1966/01	100	5.00E+14	3.00E+14
	SM 3	1961/01	100	5.00E+15	2.00E+15
	BOR 60	1969/12	60		
Sweden	R-2	1960/05	50	4.00E+14	4.00E+14
USA	ATR	1967/07	250	8.50E+14	1.80E+14
	HFIR	1965/08	85	2.10E+15	1.00E+15
	MURR	1966/10	10	6.00E+14	1.00E+14
	(FFTF)	1980/02			
South Africa	SAFARI	1965/03	20	2.40E+14	2.80E+14

* = Has not started up/not critical yet.

Table 7. Overview of today's fuel/target fabrication, reprocessing and PIE R&D facilities.

Country	Organization	Facility	Current Activities in Relation to Advanced Nuclear Energy System Development
Belgium	SCK-CEN	LHMA	The principal activity is related to R&D on the behavior under irradiation of nuclear fuel: both current and new type of fuels (UO_2 fuel, gadolinium bearing fuel, MOX fuel, niobium-doped cladding material, ...), up to high burnup, and both under normal and (simulated) transient operating conditions.
	Belgonucléaire	P0 Plant	The P0 plant produced half of the SNR; 300 first core fuel elements have been fabricated in the past. In the factory, the MIMAS or MIGRA processes can be used for the fabrication of fast reactor fuel.
Canada	AECL	Chalk River	Development of CANDU/CANFLEX fuel.
France	CEA	ATALANTE	This facility has developed a set of tools to manufacture americium-based ceramics where these tools may be used for other actinides also.
			Research on hydrochemical and pyrochemical reprocessing technologies.
		LECA/STAR Cadarache	PIE of fuel rods, targets
Germany	ITU (JRC-EC)	Minor Actinide Lab	The lab consists of two glovebox chains that are being built to fabricate fuels and targets containing significant quantities of minor actinides such as americium and curium.
		Pyro-Processing Lab	
			Also research on hydrochemical and pyrochemical reprocessing technologies.
			PIE of fuel rods, targets, and structural materials
India	IGCAR		Facilities for research on electro-oxidative dissolution, electrolytic partitioning and denitration, removal of dissolved and degraded organic materials.
Japan	JNC	*O-arai Engineering Center*	
		Fuels Monitoring Facility (FMF)	Performs PIE of the fuels and materials irradiated in "JOYO" and Overseas Reactors.
		Alpha-Gamma Facility (AGF)	
		Material Monitoring Facility (MMF)	
		Tokai Works	
		Recycle Equipment Test Facility (RETF), currently Chemical Processing Facility (CPF)	The RETF is now under construction at Tokai to test new types of equipment and processes on an engineering scale.
		Fuel Manufacturing Plant	The plant capacity is 10 tons a year for Fugen and another 5 tons per year for Monju and Joyo

Table 7. (continued).

Country	Organization	Facility	Current Activities in Relation to Advanced Nuclear Energy System Development
	JAERI	NUCEF Nuclear Fuel Cycle Safety Engineering Research Facility	Safety research on fuel reprocessing facilities and radioactive waste management.
			A part of NUCEF, i.e., BECKY (Back-End Fuel Cycle Key Elements Research Facility), alpha-gamma and gloveboxes are available for advanced fuel reprocessing and TRU waste management research (in collaboration with CRIEPI).
Netherlands	ECN	HCL	PIE of fuel
Norway	IFE	Halden Reactor Labs	PIE and re-fabrication of fuel rods
Switzerland	PSI	Laboratory for Materials Behaviour	Preparing and testing an Y_2O_3 stabilized uranium free $(Pu,Zr)O_2$ fuel for deployment in LWRs.
		Actinide Chemistry Lab	Developing a (Pu, Zr)N fuel matrix as a precursor for an (Am,Zr)N matrix for Americium incineration.
			Cooperating with an industrial partner to improve the flow sheet for MOX production.
			Studying the irradiation behavior of high burnup fuel by PIE.
			Modeling the fuel irradiation behavior with thermo-mechanical and neutronics codes.
Russia	RIAR		Facilities for research on electrochemical process (oxide process) and the so-called DOVITA process. Also research on fluoride volatilization.
Korea	KAERI	PIEF (Post Irradiation Examination Facility)	PIEF consists of three pools, four heavy concrete cells, and two lead cells. It provides adequate support services of post irradiation examination (PIE) for nuclear power reactor fuels as well as for experiments associated with reactor safety studies, and fuel design and fabrication improvements. Capability of the PIEF includes in-pool nondestructive examination, in-cell nondestructive examination, metallographic examination, and chemical composition analyses of irradiated nuclear fuels. The PIE facility was put into service for nuclear power plant fuel from 1987.
		IMEF (Irradiated Materials Examination Facility)	IMEF for supporting of research and development of nuclear fuels and materials has been operating since 1994. The facility consists of three hot cell rows, i.e., fuel test row, material test row, and multiple-use row in channel shape. Hot cells for fuel test are used for dismantling capsules and fuel bundles and examination of irradiated fuels. Hot cells for material tests are used for measurement of mechanical and physical properties of fuel and materials. The hot cell for multiple-use is currently used for the DUPIC fuel development.

Table 7. (continued).

Country	Organization	Facility	Current Activities in Relation to Advanced Nuclear Energy System Development
UK	BNFL	Berkeley Centre Sellafield Centre Technology Centre	BNFL's new Technology Centre (BTC) will also be provided with fuel fabrication facilities similar to those in their older B33 Lab F. However, rod length is limited to 1 m and there are no assembly building facilities. BTC may be able to fabricate a small number of pellets and short rods for experimental purposes, but it would not be possible to fabricate reload quantities of fuel. The Sellafield center also performs research on hydrochemical and pyrochemical reprocessing techniques.
	UKAEA	Windscale Buildings B13/B14	PIE of fuel and reactor components.
	AEA Technologies	Harwell Radiochemistry Complex	Dissolution of high burn-up fuels. Development of solvent extraction processes for fuel reprocessing. Measurement of basic solvent extraction data.
USA	INEEL	Remote Analytical Laboratory TRA Hot Cell Facility	
	ANL-W	Fuel Conditioning Facility (FCF)	FCF is where electrometallurgical treatment is performed. It consists of two heavily shielded hot cells, one with an air atmosphere and one with an inert argon atmosphere. Work can be done on extremely radioactive materials with no radiation exposure to workers or visitors.
		Hot Fuel Examination Facility (HFEF)	HFEF is a large, highly versatile hot cell facility. It features an air cell and argon cell, superb overhead crane capacity, large floor space with high ceiling clearance, and versatile access ports.
	ORNL	Irradiated Materials Examination and Testing (IMET) Irradiated Fuels Examination Laboratory	

Table 8. Overview of today's nuclear data measurement and integral experiment facilities.

Domain	Country	Organization	Facility	Current Activities in relation to advanced nuclear energy system development
High-Resolution Cross Section Measurements				
	Belgium	EC- IRMM	GELINA	Measurements from subthermal to about 20 MeV.
	Switzerland	CERN	n-TOF	Measurements from 0.1–20 MeV
	Netherlands	KVI	AGOR Cyclotron	
	USA	ORNL	ORELA	Measurements from eV to MeV
		LANL	WNR	Measurement from MeV to TeV region
			MLNSC	Below about 1 keV
		Rensselaer Polytechnic Institute	Gaertner Linac Laboratory	Up to several hundred eV
	Japan	Kyoto University		Linac white neutron source
		Tokyo Institute of Technology		Peletron
				Up to keV range
Point Energy Cross-section Measurements				
	Belgium	EC-IRMM	Van de Graaff	
	Germany	PTB Braunschweig	Van de Graaff	
		KfA Jülich	Cyclotron	Measurement of cross sections at MeV energies by radiochemical analysis of the activation products.
	UK	Birmingham University	Dynamitron	Monoenergetic measurements of delayed neutron yields
	USA	University of Lowell	Van de Graaff	
		Ohio University	Van de Graaff	
		Triangle University Nuclear Lab	Van de Graaff	
	Japan	Tohoku University	4.5 MV dynamitron	
		JAERI	Tandem	
		Kyoto University Reactor	KUR	
	Sweden	Uppsala University	Department of Neutron Research	

Table 8. (continued).

Domain	Country	Organization	Facility	Current Activities in relation to advanced nuclear energy system development
14-MeV neutron sources				
	Japan	Osaka University	Oktavian	
		JAERI	FNS	
	Austria	University of Vienna	IRK	
	Italy	INFN Frascati	LNF	
Broad Spectrum Sources and Critical Assemblies (CA)				
	Argentina	CNEA	RA8	
	Brazil	IPEN	IPEN MB01	
	Belgium	SCK-CEN	VENUS (CA)	Thermal critical assembly
	China		ZPR Fast	
	France	CEA	EOLE (CA)	Thermal critical assembly
			MASURCA (CA)	Fast critical assembly
			MINERVE (CA)	Thermal critical assembly
	Switzerland	PSI	PROTEUS (CA)	
		Swiss Federal Institute of Technology, Lausanne	CROCUS (CA)	
	Japan	JAERI	VHTCR	
			FCA	Fast Critical Assembly
			TCA	Thermal Critical Assembly
		Toshiba	NCA	
		Kyoto University	KUCA	
	USA	LANL - LACEF	BIG TEN	
			GODIVA	
			FLATTOP	
			COMET	
			HONEYCOMB	
			MARS	
		Rensselaer Polytechnic Institute	RCF	

Table 8. (continued).

Domain	Country	Organization	Facility	Current Activities in relation to advanced nuclear energy system development
Yield and Decay Data of Fission Products, and Decay Heat Measurements				
	Sweden	Uppsala University	Studsvik laboratory	Measurement of radioactive decay data and fission product yields for short-lived products and the characteristics of delayed neutron emission.
	France	ILL	Lohengrin	Recoil mass analyses
			Cosi-Fan-Tutte	
	Switzerland	CERN	ISOLDE	Separators
	Finland		IGISOL	
	Germany	GSI Darmstadt	ISOL	Separator
			FRS	
	USA	Lowell University		
	Japan	Kyoto University	KUR-ISOL	
		JAERI	JAERI-ISOL	
			TIARA-ISOL	

Other information on these facilities has also been reported in literature (6, 7, 8, 9) and the reader may find additional information in (10, 11).

4.1.1.1   Knowledge Preservation, Codes, and Databases. This note has not addressed the issues relating to knowledge preservation and transfer, calculation codes, and the continuing need of availability of databases. Work on knowledge preservation is undertaken in most of the nuclear research labs and on an international level by IAEA and OECD/NEA (e.g., Preservation of Integral Experiments Data Needs, Methods, Formats).

Databases relevant to the subject of advanced fuel cycles are related to:

1. *Nuclear Data*: as most of the fuel cycles make use of high-actinide-loaded fuels in thermal and fast spectra, a continuous need exists to complement the available nuclear data cross-section sets. The High-Priority Request List is therefore of high importance.

2. *Fuel material properties*: Many databases on properties of materials exist and some of them contain information about relevant actinide compounds. Examples are the NEA Thermochemical Database Project, the INSC Material Property database of ANL or the IVTAN Thermochemical database of the Russian Academy of Sciences. However, with respect to the compounds of transuranium elements, these databases are incomplete, as limited experimental data for these compounds exists. Also, the scope of most of the databases is not on fuel technology, and as a result, not all relevant properties are included. Because of these reasons, ITU has started an activity on an actinide/lanthanide database.

3. *Irradiation data*: The irradiation data on advanced fuel forms have not been collected systematically in databases. Fundamental data for pyrochemistry: Data on Gibbs Energies of formation of actinide, lanthanide, and fission product chlorides are available in several databases. In this respect, it should be mentioned that the actinide chlorides (U, Np, Pu, Am) have been carefully evaluated for the NEA-TDP. An assessment of the lanthanide trichlorides is also available.

4. *Separation data*: There is a significant amount of experimentally determined distribution coefficients for pyrochemical separation in chloride salts. However, they have not been collected systematically in databases.

4.1.1.2   Perceived Critical Needs for R&D Infrastructure.

4.1.1.2.1   Nuclear Data and Reactor Physics Experiments—Current facilities for nuclear data measurements and reactor physics experiments (e.g., critical assemblies) appear to be adequate world-wide and those should be able to respond to the needs of existing or proposed advanced systems. However, phasing out additional facilities would lead to a shortage of expertise and lack of independent measurement and review capabilities. International collaboration remains of primordial importance to keep these facilities running.

4.1.1.2.2   Fuel/Target fabrication—Limited possibilities exist in the world to handle and fabricate fuels/targets containing a significant amount of minor actinides, especially for curium. Let alone the availability of these facilities, one should also consider the availability of material (e.g., at present americium and curium isotopes are available in small quantities (grams or tens of grams), the appropriate waste treatment (The treatment of storage of waste containing americium or curium is far from trivial, especially concerning liquid waste. Moreover, it will be very expensive), and the transport of these fuels/targets between different facilities.

Only CEA, JNC, and Argonne National Laboratory-West currently seem to own integrated R&D infrastructure covering not only irradiation reactors but also the needed fabrication, reprocessing, and PIE facilities in rather close vicinity to each other. However, as these facilities are integrated in a set of R&D programs and some of these facilities are unique, a vulnerable situation exists since not all the Gen-IV R&D needs might be covered by these facilities only. Additional problems occur in relation to research on defective advanced fuel/targets and when the investigations are not limited to NDT alone.

4.1.1.2.3   Irradiation Facilities. The most urgent need seems to occur in the availability of fast spectrum R&D irradiation facilities. After 2005, only fast spectrum loops might be made available in some of the existing material test reactors.

4.2   Japan

The Fast Breeder Reactor (FBR) cycle system development in Japan has been promoted as a national project by Japan Nuclear Cycle Development Institute (JNC) incorporating the R&D activities of the governmental and private sectors. The JNC has three major facilities related to the FBR cycle system: (1) Joyo, (2) Monju, and (3) Chemical Processing Facility.

4.2.1   Joyo

Joyo, experimental sodium-cooled fast reactor at O-Arai Engineering Center, has operated since 1978. It has three main objectives: (1) advancement of technology through operation and experiment, (2) conducting irradiation tests on fuel and materials and Validation of innovative technology for

development of future FBRs. Its irradiation capability has been improved through Mark-I, Mark-II and Mark-III corresponding to the requirement of high performance fuels for FBRs as shown in Figure 9 and Figure 10. The initial criticality of the Mark III is scheduled in summer of 2003, and the rated power operation will start late in 2003. It has the irradiation plan shown in Figure 11.

Items		MK-I	MK-II	MK-III
Reactor Thermal Output	(MWt)	50/75	100	140
Max. Number of Driver Fuel S/A		82	67	85
Max. Number of Test Fuel S/A		0	9	21
Core Diameter	(cm)	80	73	80
Core Height	(cm)	60	55	50
^{235}U Enrichment	(wt%)	~23	~18	~18
Pu Content Total	(wt%)	~18	<30	<30
Fissile	(wt%)	–	~20	~16/21*
Max. Linear Heat Rate	(W/cm)	320	400	420
Neutron Flux Total	(n/cm^2•s)	3.2×10^{15}	4.9×10^{15}	5.7×10^{15}
Fast (>0.1MeV)	(n/cm^2•s)	2.2×10^{15}	3.2×10^{15}	4.0×10^{15}
Max. Burn-up (Pin Average)	(GWd/t)	42	75	90
Primary Coolant System Flow Rate	(t/h)	2,200	2,200	2,700
	(ºC)	370	370	350
	(ºC)	435/470	500	500
Blanket/Reflector/Shielding		Blanket/SUS	SUS/SUS	SUS/B$_4$C

*) Inner/Outer Core
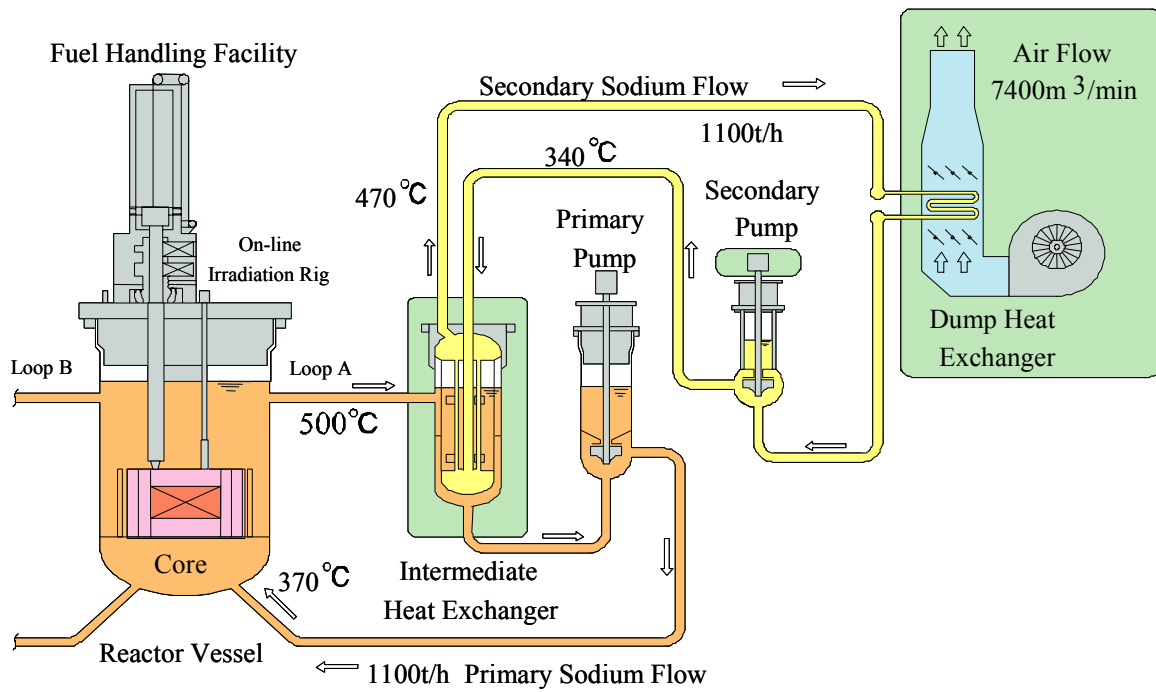
Figure 9. JOYO basic specification.
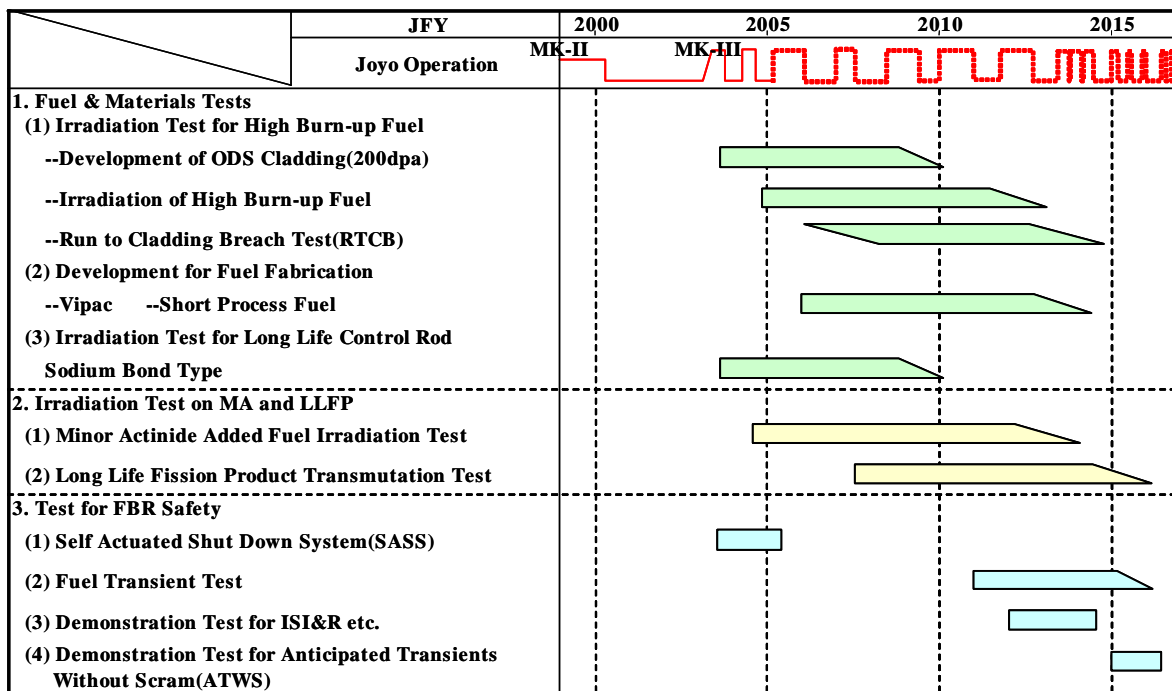
Figure 10. Joyo heat transport system.

Figure 11. Joyo irradiation test program.

Monju

The construction of Monju, prototype sodium cooled fast breeder reactor was completed in April 1991 and started pre-operation test. It has stopped operation since the sodium leakage on the secondary loop happened in December 1995. Now, safety licensing for the plant modification Monju is under examination. After its approval, modification works will be performed and it will restart around in 2005. The plant outline is described in Table 9 and Figure 12.

Table 9. MONJU plant-basic specifications.

|	
Reactor type	Fast breeder reactor
Cooling System	Sodium cooled (loop-type)
Thermal output	714 MW
Electrical output	280 MW
Fuel	Plutonium-Uranium mixed oxide
Core Dimensions:	
Equivalent diameter	180 cm (approx.)
Height	93 cm
Volume	2,340 liters
Plutonium enrichment (% Pu fiss.) Inner / Outer	16 / 21
Fuel Mass:	
Core (U+Pu metal)	5.9 t
Blanket (U metal)	17.5 t
Average burn-up	80 000 MWd/t (approx.)
Cladding tube dimensions Outer diameter / thickness	6.5 mm / 0.47 mm
Cladding tube material	SUS 316
Power density	275 kW / liter
Blanket thickness Upper / lower / radial	30 / 35 / 30 cm
Breeding ratio	1.2 approx.
Primary sodium temperature Reactor inlet / outlet	397° / 529°C
Secondary sodium temperature IHX inlet / outlet	325° / 505°C
Number of loops	3
Reactor vessel dimensions Height / diameter	18 / 7 m
Steam Pressure Before main stop valve	127 kg/cm^2
Interval between refueling	6 months (approx.)
Refueling system	Single rotating plug with fixed arm fuel handling machine
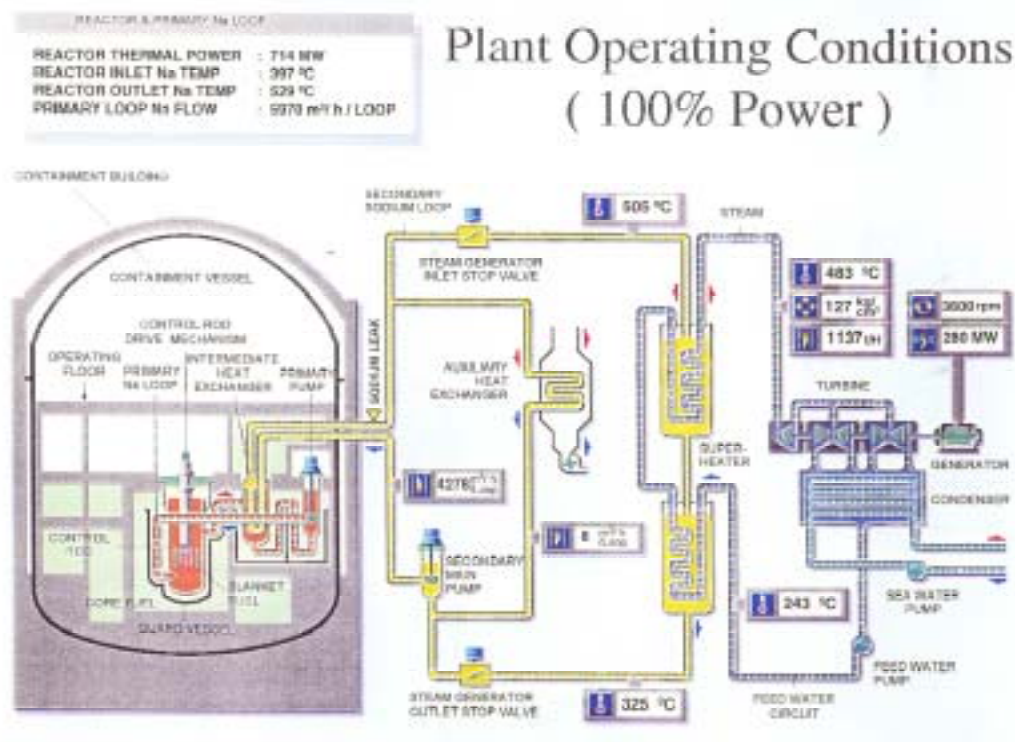
Figure 12. Plant Operating Conditions at 100% Power.

4.2.2   Chemical Processing Facility

The chemical processing facility (CPF) at Tokai-Works of JNC is used as the laboratory for the development of spent fuel reprocessing technology. It was constructed in 1980. In the first stage, the PUREX process to the Fast Reactor spent fuel was demonstrated there. Based on these experiences, JNC proposes an advanced aqueous method aiming at improving the economy and the proliferation resistance and reducing the environment burden by introducing innovative technologies. The renovation of the CPF was started from 1996 to develop new technologies and it will be finish by the middle of 2002. The glove boxes for the development of non-aqueous reprocessing technology also will be installed there under the collaboration study with CRIEPI.

The renovation consists of major activities such as remodeling of the CA-3 cell at the operation room A and installation of globe boxes and analytical equipments to the laboratory C and so on as shown in Figure 13 and 14. The CA-3 cell is the main cell in which important equipments such as a dissolver, a clarifier and extractors are installed for carrying out the hot test using the irradiated Fast Reactor fuel as shown in Figure 15. For example, concerning quantity permitted, up to 10g of Pu+U235 or 4.32MBq of radioactive substance can be used for the basic experiment and 220g of Pu+U235 for storage. The laboratory C is a new room for the development of non-aqueous reprocessing technology where up to 220g of Pu+U235 can be used as shown in Figure 16. The renovation works for the CA-3 was finished in March 2002 and the laboratory C will be finished by June 2002.
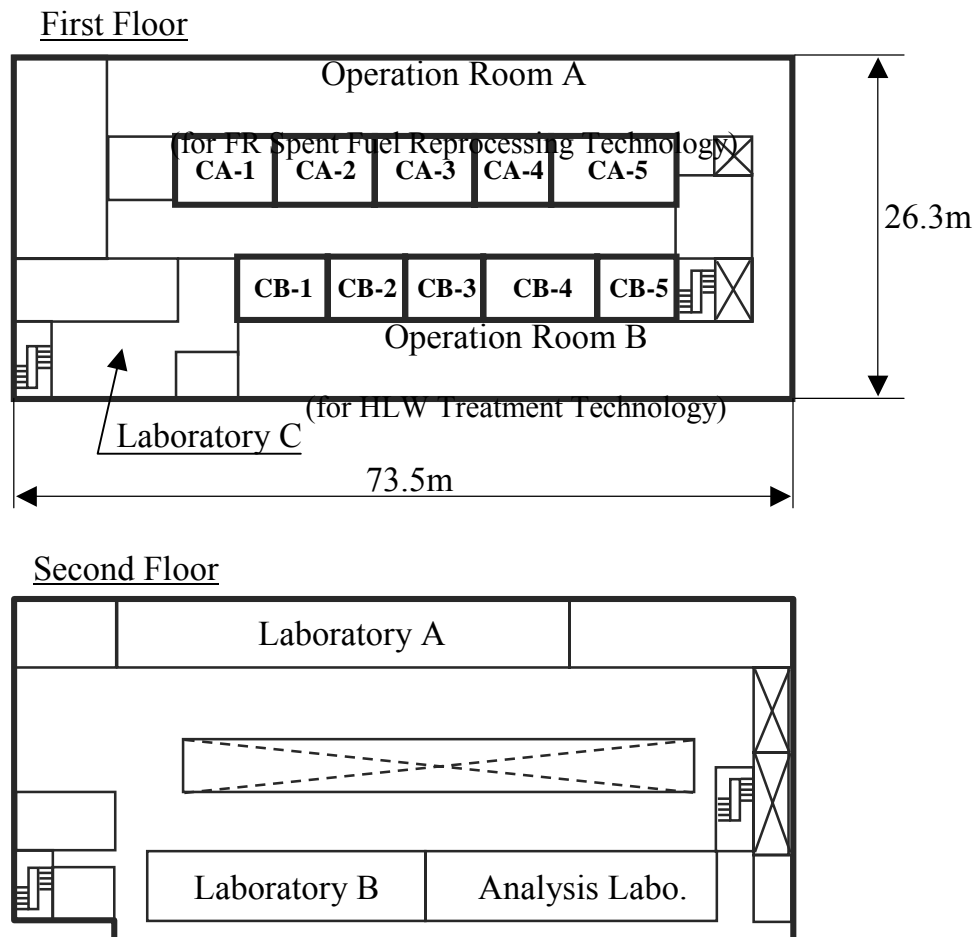
Figure 13. Chemical Processing Facility (CPF).

First Floor

| Operation Room A |
| (for FR Spent Fuel Reprocessing Technology) |

CA-1 CA-2 CA-3 CA-4 CA-5

CB-1 CB-2 CB-3 CB-4 CB-5

Operation Room B

(for HLW Treatment Technology)

Laboratory C

26.3m

73.5m

Second Floor

Laboratory A

Laboratory B   Analysis Labo.

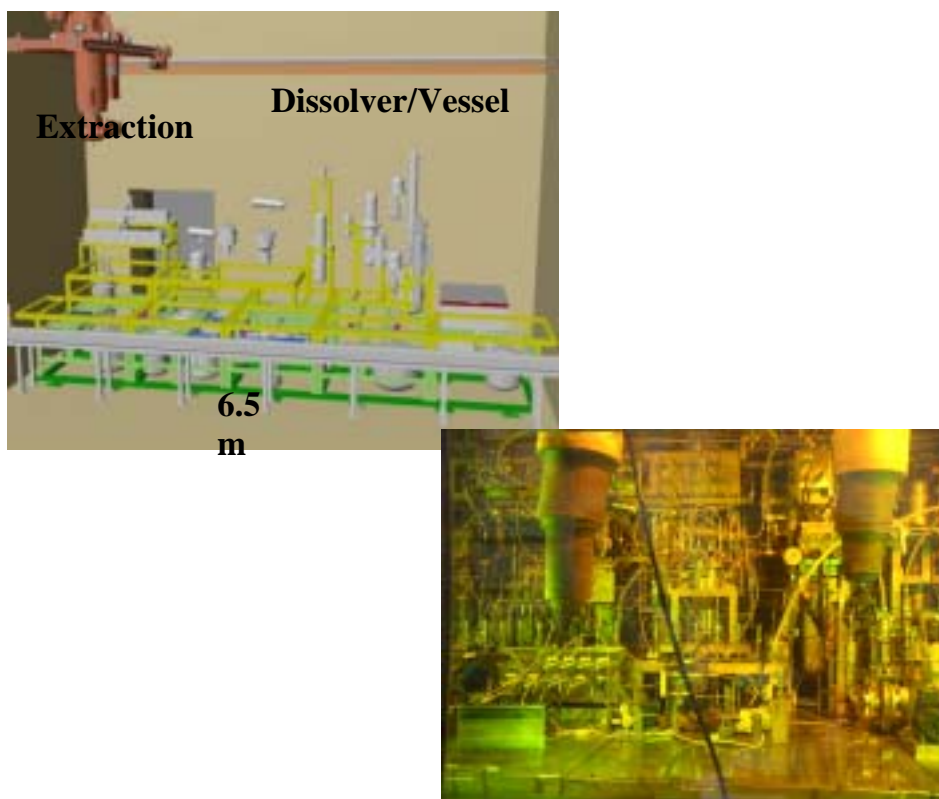Figure 14. Plane sketch of CPF.
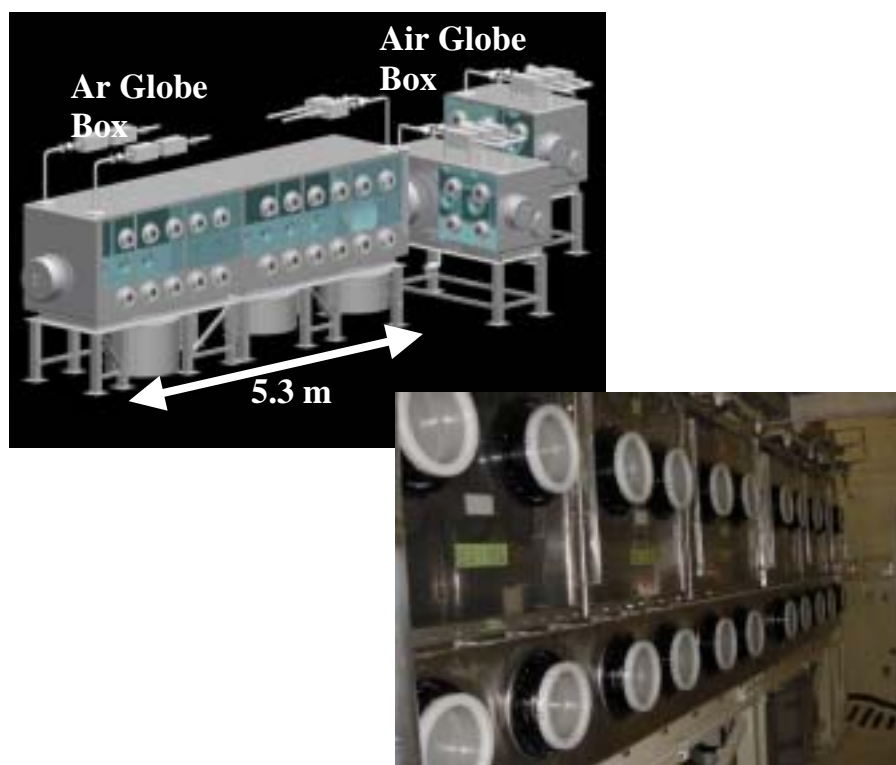
Figure 15. Renovation of CA-3 cell.

Figure 16. Installation of glove boxes in Laboratory C.

4.3   France

4.3.1   Fuel Fabrication (Test Elements)

→ LECA (cadarache center): fabrication at laboratory scale of test elements, which may include plutonium or minor actinides.

→ New pilot fabrication line for coated particles (Cadarache and Grenoble centers): kernel fabrication (UO2) and coatings (buffer + SiC + PyC) – Under design.

4.3.2   Fuel Irradiation

→ OSIRIS reactor – 70 MWth – (Saclay center): experimental reactor (pool type) with thermal flux of 2.5 10E14 n/cm^2.sec.

→ PHENIX reactor – 230 Mwe – (Marcoule center): fast neutron reactor (sodium cooled) with a flux of 10E15 n/cm^2.sec

→ RJH (Cadarache center): new reactor under design for future irradiation experiments (will be in operation by 2010)

4.3.3   Irradiated Fuel and Material Examination

→ LECA (Cadarache center): dedicated to non-destructive and destructive analysis of irradiated fuels with the help of numerous devices and methods

→ PELECI (Saclay center): will be operational by 2003. Dedicated to irradiated material tests and analysis.

4.3.4   Fuel Cycle Back-End (Reprocessing, Waste Treatment, and Long Term Behaviour)

→ ATALANTE (Marcoule center): versatile hot cells for any R&D on fuel reprocessing and experimental studies on waste treatment, conditioning and long-term behavior.

4.3.5   Core physic (Qualification of Neutronic Calculations)

→ EOLE and MINERVE (Cadarache center): critical mock up – pool type – dedicated to measurements of all neutronic parameters using in particular "oscillation methods" (cross sections, "buckling", kinetic parameters, temperature coefficients, etc…).

→ MASURCA (Cadarache center): critical mock up in air – Same as above but for fast spectrums.

4.3.6   Fuel Modeling (Computer Codes)

→ Coupled neutronic

→ Fuel performance does for HTR fuels (under development)

5. TECHNOLOGY GAP AND RECOMMENDED R&D SPREADSHEETS FOR VIABILITY-PHASE CROSSCUTTING R&D

FCCG-(FCCG Top Ranked R&D for Viability Phase)
R&D Scope for Concept (   )
Version (July 1, 2002)

Sub-System	Cost	Technical gap/issue			R&D items				
		Brief Description of Gap/Issue	Signific. of Gap (a)	Current TRL (b)	Activity Label	Brief Description of R&D Activity	Priority (c)	Time (d)	Estimated Cost Range (Million USD)
Fuel Cycle	$1000 K/y *5y	Once-Thru, Open Future Integrated Fuel Cycle	V	2		System Study of Multi Purpose Cask Strategy for Once-thru Waste Management Design Option Trade Study for Variable Heat Removal Cask Design Design Option Trade Study of Ventilated Repository Concepts	2	M	

	$3000 K/y *5y	Cs/Sr Heat Management Strategy Study in Recycle Fuel Cycles	V	2		Systems Study Near-Tem Heat Management Options and Effects
						  - Options for early separation (pre recycle)
						  - Options for mid separations (pre vitrification)
						  - Options for late separation (pre emplacement)
						  - Options for separate design
							2	M	

	$200 K/y x 5 U's x 4y = $4,000K
	
	$1000 K/y * 8y's = $8,000K
	Total $12,000K
		Aqueous Group Extractant Development	V	1		Extractant Molecule Design Campaign
						  - Combinatorial Chemistry; Molecular Dynamics
						  - Screening of Candidates
						
						Surrogate Bench Testing
						
						Hot Cell Testing
							2	M	

	$1,000 K/y x 5y Systems $5,000 K/y x 5y Testing Total = $30,000K	Systems Evaluation of Homogeneous vs Heterogeneous Recylce of Minor Actinides	V	2		Full scope life cycle evaluation of cost, dose, infrastructure implications of homogeneous and heterogeneous Cm management strategies Cm target fabrication option sutdy screening and option selection Hot cell testing	1	M	
	$1,000 K/y * 5y plus Recyle plus Refrabrication Totals	Recovery Fraction Optimization of Pyro Recycle	V	3		Systems evaluation of cost/payoff relationships of recovery fraction/ repository performance Engineering scale optimization of recyle recovery fraction Engineering scale optimization of remote refabrication loss fraction	1	L	
	500 K/y *5y	Ongoing Systems Study Evaluations of Integrated Fuel Cycle/Integrated Waste Management	V	2		Mass Flow, Waste Arisings, Heat Loads, Shipping, etc. Economics Evaluations	1	M	

a Indicate relevance of technology gap: V = concept viability, P = performance, O = design optimization
b Indicate technical readiness level (1, 2, 3, 4, or 5); see EMG Final Screening Document
c Indicate priority of R&D activity:
    1 = critical (needed to resolve a key feasibility or viability issue)
    2 = essential (needed to reach a minimum targeted level of performance, or to resolve key technology or performance uncertainties)
    3 = important (needed to enhance performance or resolve the choice between viable technical options)
d Indicate time required to perform R&D: S = short (<2y), M = medium (2-5y), L = long (5-10y), VL = very long (>10y)

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